

**LETTER REPORT  
SURVEY AND RECOMMENDATIONS FOR POTENTIAL  
DRYING AND SEALING ISSUES WITH TAD DESIGNS**

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*Prepared by*

**T. Wilt**

**Center For Nuclear Waste Regulatory Analyses  
San Antonio, Texas**

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## QUALITY OF DATA ANALYSIS AND CODE DEVELOPMENT

**DATA:** No CNWRA-generated data are referenced in this report. Data from other sources are included with references to their source. Sources of these non-CNWRA data should be consulted for determining levels of quality assurance.

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# 1 DRAFT LETTER REPORT SURVEY OF POTENTIAL DRYING AND SEALING ISSUES FOR TRANSPORTATION, AGING, AND DISPOSAL CANISTER DESIGN

## 1.1 Introduction

Under the Nuclear Waste Policy Act of 1982, the U.S. Department of Energy (DOE) is responsible for disposal of civilian spent nuclear fuel in a deep geologic repository. The DOE is also responsible for any interim storage of the spent nuclear fuel prior to disposal as well as its transportation in connection with disposal or storage.

In response to these requirements, DOE began evaluating the fabrication of a standardized container system that would provide for storage, transportation, and possible disposal of spent nuclear fuel.

### 1.1.1 Background

The initial dual-purpose canister design began in April 1995 when a contract was awarded to Westinghouse Electric Corporation for both large and small capacity storage canisters, a prototype transportation overpack, and welding and handling equipment. In November 1995, as a result of budgetary constraints, DOE decided to terminate work to develop the dual-purpose canister system. All work that was performed was considered to be nonproprietary, and the design packages and Topical Safety Analysis Reports were made available to private industry. Westinghouse elected to continue efforts to have the large U.S. Nuclear Regulatory Commission (NRC) dual-purpose canister (DOE, 1997). In addition, other companies in the private sector then continued the development of the dual-purpose canister concept. For example, dual-purpose canisters have been utilized as part of the HOLTEC HI-STORM™ and S100 Storage system. Most recently, companies such as HOLTEC International, Transnuclear Inc., NAC International, and EnergySolutions Spent Fuel Division Inc. have submitted proof-of-concept designs for the proposed transportation, aging, and disposal (TAD)<sup>1</sup> canister based upon the experience gained through existing dual-purpose canister designs (DOE, 2007a).

In existing designs, the dual-purpose canister is a sealed canister consisting of an internal fuel basket. There are different dual-purpose canisters with different fuel basket geometries for storing pressure water reactor or boiling water reactor fuel. This same functionality will be inherent in the proposed TAD. The canister shell enclosing the fuel basket is considered the confinement<sup>2</sup> boundary, which is defined as the sealed, cylindrical enclosure of the dual-purpose canister shell that is welded to a solid baseplate (bottom lid). A top lid is welded around the top circumference of the shell wall. Access ports used for drying and inerting operations are sealed by cover plates, which are welded to the lid. Finally, a closure ring is welded to the lid and canister shell providing redundant sealing. Similar sealing operations will be a key component in the containment<sup>3</sup> function of the proposed TAD. As required in the current proposed TAD specifications, the final closure welds must meet Interim Staff Guidance

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<sup>1</sup>Transportation, aging, and disposal is referenced frequently in this report. The acronym TAD will be used.

<sup>2</sup>*Confinement* is the system, including ventilation, that acts as a barrier between areas containing radioactive substances and the environment (as defined in 10 CFR Part 72).

<sup>3</sup>*Containment* is the assembly of components of the packaging intended to retain the radioactive material during transport (defined in 10 CFR Part 71).

(ISG)<sup>4</sup>–18 (NRC, 2003b) to ensure that there will be no credible leakage in containment and confinement. As part of the sealing specification (DOE, 2007a), the proposed TAD must be tested to demonstrate a leaktight containment boundary and the leak testing process shall conform to ANSI N14.5-97. As will be discussed here, such redundant sealing is a requirement given in 10 CFR Part 72.

The objective of the drying process is to remove any moisture contained within the canister and the spent fuel rod cladding. Water has a detrimental effect on the spent fuel rod cladding; hydrogen embrittlement and oxidation degrade the performance of the cladding. Current proposed TAD requirements specify that the draining and drying operations be carried out according to NUREG–1536 (NRC, 1997). The two common drying processes are vacuum and forced gas flow. Vacuum drying relies on reduced pressure relative to the atmosphere to evaporate the moisture and the vacuum to remove the moist air from the canister. The forced gas flow utilizes an inert gas, such as helium, at a reduced pressure to extract the moisture. In this process, the helium may also be preheated to increase moisture removal. Most importantly, the temperature during the drying process must remain at such a level that no high thermal strains, which could rupture the fuel cladding, are generated. Therefore, cladding temperature limits may dictate the type of drying process (NRC, 1997). At this time, the proposed TAD specifications do not specify the type of drying process and individual vendors will propose the drying method. After draining, the canister will be backfilled with helium to provide an inert atmosphere inside the canister.

### **1.1.2 Objectives and Outline**

The objective of this report is to review the current understanding of drying and sealing issues when applied to current spent fuel canisters. Through this review, insights into the requirements for the proposed TAD will be investigated.

The outline of this report is as follows: Section 1.2 briefly discusses the 10 CFR Parts 71 and 72. Section 1.3 discusses the applicable NUREGs and NRC ISGs as they pertain to drying and sealing of spent fuel canisters and issues with respect to spent fuel cladding integrity. Section 1.4 presents a preliminary discussion of how the TAD will be required to satisfy 10 CFR Part 63.

Section 2.1 addresses the forms of sealing designs used in current spent fuel canisters. Section 2.2 discusses drying processes, and Section 2.3 discusses leak testing criteria and the different procedures for performing the leak test. Finally, Section 2.4 presents a summary and conclusions based upon the topics discussed in Section 2. Appendix A provides a report on supplemental drying/sealing guidance needs for transportation, aging, and disposal canister design.

## **1.2 Title 10 Code of Federal Regulations: Parts 71 and 72**

NRC regulations in 10 CFR Parts 71 and 72 govern the transportation and storage of spent nuclear fuel, respectively. Together, these regulations address the following safety objectives: (i) ensure that the doses are less than the limits prescribed in the regulations; (ii) maintain subcriticality under all credible conditions of storage and transportation; and (iii) ensure there is adequate confinement, containment, and shielding of the spent fuel during storage and transportation (NRC, 2003a).

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<sup>4</sup>Interim Staff Guidance is referenced frequently in this report. The acronym ISG will be used.

The transportation package must provide containment such that there is no loss (or dispersal) of the radioactive contents as per 10 CFR 71.51(a)(1), and the package must provide shielding so that surface radiation does not increase under normal conditions of transport as per 10 CFR 71.51(a)(2). Likewise, 10 CFR 71.51(a)(2) for hypothetical accident conditions specifies that the package must provide adequate containment and shielding to meet specific limits on any increase of external radiation under normal conditions of transport or release of material. The requirements of 10 CFR 72.55 include criticality safety requirements for a single transportation package. As given in 10 CFR 71.55(d), under normal conditions of transport, the package must be designed such that the contents remain subcritical and the geometric form of the contents will not be substantially altered when the transportation package is subjected to a number of possible dynamic events (i.e., as given in 10 CFR 71.71: vibration, a free and corner drop, possible penetration due to a falling object). The structural performance of the package must also be sufficient under hypothetical accident conditions (10 CFR 71.73) when subjected to dynamic events (specifically a free drop, puncture, and fully engulfing fire for 30 minutes) such that the contents remain subcritical as per 10 CFR 71.55(e). The fuel reconfiguration (geometry) after a hypothetical accident condition may not be known (e.g., in a sealed proposed TAD canister, making it difficult to assess the condition of the fuel). The licensee will have to obtain data to show that the fuel remains subcritical. Considering these requirements, the package must not only be structurally robust, but the transportation package must also be sufficiently sealed to prevent a breach of containment in the event of transportation accidents. These requirements are especially important when considering the transportation of high burnup fuel in which the structural integrity of the spent nuclear fuel rods must be evaluated for various conditions of transport.

As discussed previously, dose limits, subcriticality, and confinement are also addressed in 10 CFR Part 72, which specifies the requirements for designing the spent fuel cask<sup>5</sup> for storage. According to 10 CFR 72.236(d), the storage cask must be able to provide shielding and confinement to satisfy the dose limits as specified in 10 CFR 72.104(a) and 72.106(b). As per 10 CFR 72.236(j), the storage cask must be inspected to confirm that its confinement capability is not affected by defects (such as cracks and pinholes) or unacceptable voids. As part of ensuring confinement, the storage cask must be designed to provide redundant sealing [10 CFR 72.236(e)]. Redundant closures of the canister (i.e., lids, seals, and welds) will satisfy the necessary confinement requirements. One additional and significant requirement of 10 CFR Part 72 regulations is the retrievability of the spent fuel from storage as defined in 10 CFR 72.122(l). Because of this requirement, the storage cask must be designed so it is compatible with removal of the spent fuel [10 CFR 72.236(m)]. Therefore, to ensure the possible removal of the spent fuel assemblies, its cladding must be preserved, as specified in 10 CFR 72.122(h)(1), to prevent possible rupture.

Cladding rupture may pose operational problems with respect to cladding removal from storage (Electric Power Research Institute, 2003). Thus, one technical issue was to resolve the mechanical properties (i.e., creep and fracture toughness) of advanced claddings. Current ISG-11 cladding temperature limits would avoid the creep and fracture concern. The condition of the fuel needs to be well characterized prior to dry storage and transportation. For example, during storage, spent fuel temperatures or the storage system backfill gas used for inerting could be periodically monitored to indicate the fuel integrity. Fission gas release (or lack of) into the cask could indicate the fuel integrity during dry storage (Electric Power Research Institute, 2003). However, for all-welded TAD canisters, monitoring is not feasible. Therefore, fuel characterization becomes important.

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<sup>5</sup>Cask refers to the entire waste-containing system.



One question that arises is whether high burnup fuel can become brittle leading to potential fracture when subjected to normal conditions of transport or hypothetical accident conditions dynamic (impact) loads (10 CFR 71.71). High burnup fuels may have some wall thinning from increased oxidation, which can affect the structural capacity of the fuel rod. Whether there is any loss of ductility during storage—which may affect the fuel rods' structural performance during transportation—also needs to be determined. Therefore, there must be a sufficient amount of material property data for high burnup fuel (e.g., cladding data for temperatures expected during transport). The structural capacity of the fuel rod can be measured in terms of its stiffness when subjected to a buckling analysis. This has implications on the package design such that g-force loads on the fuel may need to be limited to prevent fuel rod damage. Hydride formation is also an important issue to be considered for high burnup fuel operations (e.g., the drying process). High burnup fuel will be further discussed in subsequent sections.

### **1.3 NUREGs and Interim Staff Guidance**

An important consideration in the transportation and storage of spent nuclear fuel is the degree to which the fuel's integrity is maintained. Spent fuel integrity is mainly influenced by (i) storage temperature, (ii) fuel rod internal pressure, and (iii) the condition of the cladding. Guidelines for maximum cladding temperature limits have been set forth in both applicable NUREGs and ISG. For example, cladding creep depends on the temperature and the time-at-temperature period. One important process that will affect the temperature of the cladding is drying of the cask. It is important that the temperature remains within limits suggested in the NRC guidance to prevent cladding degradation.

The survey of the ISGs and the Safety Review Plans represents the current staff guidance that has been published pertaining to storage and transportation of commercial spent nuclear fuel. However, guidance in these areas, especially for high burnup fuels, continues to evolve as new information is obtained regarding material cladding integrity and design requirements for storage and transportation casks.

Most recently, Argonne National Laboratory (2007) proposed a test program for high burnup spent nuclear fuel cladding integrity. The results of this program will be used by spent fuel cask vendors for their transportation license applications and by the NRC for the review of these applications. The test plan will involve obtaining axial tensile properties and impact resistance data as NRC requested. This data will be used to assess the cladding performance during cask transportation accidents following drying and long-term storage. The thermal and circumferential stress history of high burnup fuel may have a substantial impact on the cladding performance when subjected to post-storage transportation accidents. During reactor operations, high burnup fuel cladding is subjected to radiation-induced hardening and hydrogen pickup. The radiation hardening leads to a decrease in ductility. Similarly, hydrogen pickup, which increases with burnup, does not result in a loss of ductility if the hydrogen stays below 1,000 ppm. If the hydride remains circumferential, hydrogen up to 700 ppm does not cause loss of ductility. The ductility loss becomes a problem when the hydride becomes radial during the drying process. The data produced from this test plan will be used along with the appropriate modeling codes to determine whether the cladding fails, the failure mode, and the extent of cladding damage. These issues are expected to apply to storage and transportation of TADs, especially for high burnup fuels.

### 1.3.1 Drying

NUREG-1536 (NRC, 1997) has accepted vacuum drying methods as outlined in Knoll and Gilbert (1987). After the cask is drained of as much water as possible, a vacuum system is used to create a subatmospheric state within the canister, which causes the moisture in the canister to evaporate. The cask is evacuated to a specified pressure of 3 torr and held constant for 30 minutes with the vacuum pump bypassed.

Maintaining this low pressure for a specified time will ensure that an adequate amount of moisture is removed. The specific operation procedures should reduce the amount of water vapor in the confinement cask to an acceptable level as required by 10 CFR 72.122(h)(1) to prevent fuel cladding degradation. Oxidation of exposed fuel could produce fuel swelling, leading to possible cladding rupture with the rates of the oxidizing reactions being highly temperature dependent. To minimize cladding oxidation during storage, the canister is placed in an inerted dry storage condition in which the canister is backfilled with an inert gas—most commonly helium. NUREG-1536 (NRC, 1997) specifies that leak testing should be performed in accordance with “American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials,” ANSI N14.5. Vacuum drying, if performed as specified, should lead to acceptable moisture levels in the cask. The amount of any water remaining in the canister or its internal components will have implications associated with its radiolytic decomposition. One key byproduct of radiolysis is the production of hydrogen, which has been shown to be detrimental to the integrity of the fuel rod cladding.

ISG-22 (NRC, 2006a) addresses possible damage to the spent fuel rods as a function of exposure to an oxidizing atmosphere. This damage takes the form of axial and/or spiral splitting of the rod. The rates of oxidation can be a function of a number of variables: burnup, the moisture content in air, cladding material, and the form of any initial defects. After the fuel has been placed in the cask and as the water is drained to a level where part of the fuel rods are exposed to air, the rods may be prone to oxidation. At the expected temperatures associated with spent fuel, the oxidation reaction can occur within a matter of hours. Therefore, ISG-22 (NRC, 2006a) recommends taking reasonable protective measures to prevent damage to the spent fuel rods.

To counter the effects of oxidation, ISG-22 (NRC, 2006a) recommends maintaining the spent fuel rods in a nonoxidizing environment by using an inert cover gas such as helium. Studies have shown that (i) for oxygen contents at atmospheres in the range of a few torr or less, the oxidation rate will decrease and (ii) for studies that have used a low partial pressure of water vapor in air, the oxidation rate has not shown any dependence on the moisture content of the air. Both of these conditions are typically present during the vacuum drying process and during helium backfill. The helium backfill provides heat transfer during storage, an inert atmosphere for long-term fuel integrity, and the means of future leakage rate testing.

One aspect of particular importance is the composition of the gas present inside the cask. Even inert cover gases will have some impurities. Impurity gases degrade the cladding by either reacting with the cladding itself or by reacting with exposed fuel. There are basically four possible sources of impurity gases: (i) residual impurities in the cover gas itself; (ii) air leakage into the cask; (iii) possible residual impurity gases remaining after evacuation is complete; and (iv) outgassing of materials (Knoll and Gilbert, 1987). As mentioned, impurities in the cask fill gas can come from air in-leakage through the container seals. However, because of the use of redundant sealing, this scenario is considered remote. Therefore, it is believed that nearly all reactive gas is introduced during initial loading and does not increase with time. The helium fill

gas itself may have impurities, but this is also considered remote if high purity (i.e., 99.995 percent) gas is used (Knoll and Gilbert, 1987).

The behavior of the impurity gases' effects on the cladding depends on two factors: (i) cladding temperature and its time dependence and (ii) the number and size of the cladding defects that could expose the fuel to oxidation. Knoll and Gilbert (1987) evaluated effects of reactive impurity gases for Zircaloy-clad fuel assemblies in sealed helium-filled canisters. The residual gas amounts depend on the drying process, which should be designed so that evacuation and backfilling prevent contamination of the cover gas and after loading. The cask should be evacuated and backfilled twice to reduce the amount of residual impurity gases. This process limits the amount of residual gases to 0.25 volume percent, which should reduce the amount of oxidants. NUREG-1536 (NRC, 1997) notes that if other drying processes (other than vacuum) are used, analyses should be conducted to verify that moisture and gas impurities will not cause cladding degradation. Forced gas drying (e.g., forced helium dehydration) is another method of canister drying. It should be expected, though, that the gas (e.g., helium) used will be of sufficient purity.

### **1.3.2 Sealing**

With respect to sealing of the cask, NUREG-1536 (NRC, 1997) addresses seal-welded cask closures. To ensure that occupational radiation exposures remain as low as reasonably achievable, using a remote welding operation for making the seal welds of the containment and confinement vessel is necessary. As part of the welding requirements, leak testing in the form of dye penetrant tests is to be performed on both the root and cover pass welds as part of the nondestructive evaluation of the closure welds. These weld tests must comply with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 2007). As will be discussed next, typical casks in use today utilize a series of welds for the shield plug, port cover plates, and sealing ring, which are used to satisfy the redundant sealing requirements [10 CFR 72.236(e)]. Helium leak testing is also used to verify the integrity of the seal welds. As part of the welding process, NUREG-1536 (NRC, 1997) specifies that possible generation of hydrogen gas due to radiolysis is considered. Any hydrogen in the cask must be purged for safety during loading (seal welds) and for any unloading operations involving welding (e.g., seal cutting).

A seal weld is defined as "any weld designed primarily to provide a specific degree of tightness against leakage" (American Welding Society, 2001). These welds contain a fluid, gas, or liquid. They not only prevent leakage out of a container, but also prevent entry of a fluid into a container where some type of harmful process may occur (e.g., corrosion). Seal welds are not specified for strength-related reasons, and caution should be used when they are specified (James F. Lincoln Arc Welding Foundation, 1999). For example, the application of a seal weld could result in a conflict of code requirement(s) (i.e., the seal weld could end up performing unintended structural applications that could result in unwanted load paths, residual stresses, and a limiting fatigue point). Also, seal welds could affect nondestructive evaluation (specifically, ultrasonic inspection) by creating alternate sound paths. Finally, they can be treated in a hurried or rushed manner by welders which could lead to quality control problems such as undersized seal (fillet) welds resulting in cracks.

The issue of canister fabrication also needs to be addressed. The design of the final closure welds of stainless steel canisters is discussed in ISG-18 (NRC, 2003b). The stainless steel canister is defined as a confinement boundary for spent fuel storage (10 CFR Part 72) and is defined as a containment boundary for transportation (10 CFR Part 71). In both cases, the

boundary prevents the leakage of the radioactive material contained within. For storage, the canister must maintain radioactive material confinement for both normal and credible off-normal accident-loading conditions. For transportation, the canister must provide containment to prevent loss or dispersal of the contents during normal conditions of transport and under hypothetical accident conditions.

The closure welds must meet the criteria set forth in ISG-15 (NRC, 2001), which recommends that the American Society of Mechanical Engineers welding code be used as the preferred code for storage cask design. All the welds of the confinement/containment canister should be full penetration welds, and the nondestructive evaluation of these welds should be volumetric.

The structural (outer) lid of the cask may be either a full penetration weld or a partial penetration groove weld. For carbon or alloy steel casks, the lid weld must be examined by ultrasonic testing or other volumetric methods. One important aspect of carbon or alloy steel welds is fracture control. Air hardening of the carbon steel during welding could have a significant adverse influence on fracture toughness in the weld area. This effect can occur in the steel material, which has a sufficient amount of carbon, causing a significant hardening of the steel that occurs during cool down in air and resulting in brittle deposits in the heat-affected zones. The temper bead welding process could be considered to optimize the properties of welds and heat-affected zone (HAZ)<sup>6</sup> in steels where postweld heat treatment is not performed or where it might be desirable, but not practical. The American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section IX (American Society of Mechanical Engineers, 2007), defines temper bead welding as: “A weld bead placed at a specific location in or at the surface of a weld for the purpose of affecting the metallurgical properties of the heat-affected zone or previously deposited weld metal.”

A temper bead is deposited to affect the properties of the HAZ or the weld metal beneath that bead that is being placed at a specific location. The temper bead improves the metallurgical properties of the HAZ or the weld metal located under the temper bead. The factors that determine exactly what microstructures exist in the HAZ after a weld bead is deposited depend mostly on two factors.

- (1) The chemical composition of the base metal. The higher the carbon equivalent, the more hardenable the steel, and the easier it is to damage during welding.
- (2) With respect to the cooling rate, faster cooling rates cause harder and more brittle microstructures in higher carbon equivalent materials. For example, thicker materials cool faster than thin ones, and welds made with lower heat input cool more rapidly than those made with high heat input. In addition, welds made on cooler base metals cool more rapidly than welds made on hotter base metals.

Other areas of the HAZ experience grain growth to form a coarse grain region. The longer this region stays above the grain growth temperature, the more grain growth there is. The larger these grains become, the more the toughness of that region deteriorates. Rapid cooling results in a smaller HAZ, smaller grain size, and less loss of toughness. Slow cooling results in more loss of toughness. To counter this effect, preheat and postweld heat treatment of the weld is important. However, fuel cladding temperature must be considered (e.g., possible overheating, thereby making welds requiring full temperature preheat and postweld treatment undesirable).

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<sup>6</sup>Heat-affected zone is referenced frequently in this section. The acronym HAZ will be used.

For stainless steel containment vessels, a multipass dye penetrant testing or a volumetric method such as ultrasonic testing may be used. For either method, the minimum detectable flaw size must be smaller than the critical flaw size, which should be calculated using American Society of Mechanical Engineers Section XI methodology “Rules for Inservice Inspection of Nuclear Power Plant Components” (American Society of Mechanical Engineers, 2007). The weld metals are to be specified according to the base material chemistry, and a small amount of overmatching in which the weld has more alloy content than the base metal is normal practice.

Note that magnetic particle testing is not discussed as an alternative testing method; this is because the TAD canister has been specified to be fabricated from a 300 series austenitic stainless steel that is nonmagnetic. The magnetic particle testing only works with a ferritic magnetic material. This method also requires a fluid containing magnetic particles, which must be applied to the weld and must be applied and removed manually.

In general, the weld strength must equal or exceed the strength of the base metal (NRC, 2001) and the weld metal strength can be defined as the yield and tensile strength of the deposited weld metal. The strength is measured from an all-weld metal tensile coupon taken from a welded joint that conforms to the applicable American Welding Society filler metal specification. The “matching” of weld metal is accomplished if it has a minimum yield and tensile strength equal to or higher than the minimum specified strength properties of the base metal. Note that the minimum properties are specified because in reality both the filler metal and the base metal have properties that are usually higher. A requirement for selecting the matching filler metal depends upon the joint type and loading condition. Appropriate matching is especially important for welded components that may be loaded into the inelastic range [e.g., the drop (impact) of a canister]. Under this type of load where yielding is expected, such deformations should be distributed throughout the base metal. For austenitic stainless steel to be used in the TAD, a small amount of “overmatching” is normally used, which means that the weld will contain more alloy content than the base metal (NRC, 2001).

As discussed in ISG-15, a weld schedule should be specified showing the base and weld material combinations that would allow a comparison of the base and weld metal properties (NRC, 2001). Determining the appropriate weld schedule depends on a number of interdependent parameters [e.g., thickness of the components to be welded, the material properties of the components (which could be different), welding current and time, electrode face diameter, feed wire rate]. In addition, to some degree, the specific welding schedule parameters specified for a particular welding operation are also dependent on the experience of the operator.

ISG-18 points out that quality assurance is a particularly important aspect of canister fabrication (noted in 10 CFR Part 71, Subpart H and 10 CFR Part 72, Subpart G). For transportation canisters, 10 CFR 71.117 ensures traceability of the materials (i.e., pedigree information) used in the fabrication to prevent the use of incorrect or defective materials and 10 CFR 71.119 specifies that there be control over the processes of welding, heat treating, and nondestructive testing. Similarly for storage canisters, 10 CFR 72.156 and 10 CFR 72.158 provide controls for material traceability and fabrication processes, respectively. ISG-18 notes that the qualification standards provided in this guidance present a sufficient alternative to the ANSI N14.5 periodic and preshipment leak-testing standards for the final closure welds. In addition, if the guidance of ISG-15 is followed regarding the final closure welds of austenitic stainless steel canisters, “the staff concludes that no undetected flaws of significant size will exist” (NRC, 2003b). Therefore, if the nondestructive testing of the welds is executed in accordance with ISG-15 recommendations, any flaws that could lead to a failure will be detected. Additionally, by

satisfying the design/qualification guidance of ISG-18 (NRC, 2003b), it is the NRC position that there can be reasonable assurance that no credible causes of leakage can occur in the final closure welds.

### **1.3.3 Cladding Integrity**

As mentioned previously in Section 1.2.1, 10 CFR Part 72 has the key requirement that the spent fuel be retrievable from storage. Because of this, it is important that the spent fuel cladding does not degrade during storage and that possible fuel rod rupture is prevented. Therefore, it is necessary to fully understand the mechanical behavior (e.g., fracture toughness, creep) of spent fuel cladding when it pertains to the transportation of high burnup fuel as per 10 CFR 71.55.

Cladding integrity during dry storage is a function of two primary specifications: selection of the cover gas and the maximum cladding temperature. Placing the spent fuel in an inert environment may be considered a conservative approach because the condition of the cladding may not be known and any defects present in the cladding would be prevented from propagating (Johnson and Gilbert, 1983). However, it is best to minimize the size and number of cladding defects because of the possibility of loss of fuel particles during handling operations. NUREG-1536 (NRC, 1997) states that the spent fuel cladding should not degrade to a point where more than 1 percent of the fuel rods fail. Fuel rod failure during dry storage is characterized by pinholes, hairline fractures, axial splits, or ductile fracture. From hot cell and laboratory tests (which were under dry storage conditions for time periods of up to 1 year), irradiated rods in an inert cover gas were subjected to temperatures from 100 to 570 °C [212 to 1,058 °F] with no apparent cladding defects (Johnson and Gilbert, 1983). Peak temperatures of 765 to 800 °C [1,409 to 1,472 °F] were required to drive fuel rods to failure (i.e., rupture) (Johnson and Gilbert, 1983). In addition, theoretical modeling has shown an acceptable peak cladding temperature of up to 430 °C [806 °F] when in inert storage. Thus, dry storage tests and theoretical modeling lead to the conclusion that Zircaloy-clad fuel can be used in dry storage under a broad range of conditions (Johnson and Gilbert, 1983). With regard to drying operations, tests have shown that when Zircaloy-clad fuel has been subjected to 570 °C [1,058 °F] temperatures over a timespan of many days, no cladding failures resulted (Johnson and Gilbert, 1983). Therefore, Johnson and Gilbert (1983) report that subjecting the cladding to the number of hours for the drying cycles will not degrade cladding integrity. However, this does depend on whether the hoop stresses in the actual fuel rod cladding are below those of the test rods. For temperature excursions near or above 400 °C [752 °F], some radiation annealing may occur, and this annealing may increase the ductility of the cladding, which tends to improve the cladding integrity (Johnson and Gilbert, 1983). Therefore, for Zircaloy fuel cladding, the temperature should be maintained below 570 °C [1,058 °F] for cask operations, such as vacuum drying of the cask; this is true for low burnup fuel only. NUREG-1617 (NRC, 2000b) cites the work of Levy, et al. (1987), who recommended using a range of temperatures for the dry storage of fuel—unlike the Johnson and Gilbert (1983) recommended limit of 380 °C [716 °F]. This range of temperatures would take into account variations in fuel design, burnup level, fuel age, and the storage cask design. Levy, et al. (1987) believe that a single-value temperature limit would lead to unnecessary conservatism.

For burnup values less than 45 GWd/MTU, NUREG-1567 (NRC, 2000a) states that there is sufficient data to support the temperature limit of 570 °C [1,058 °F]. ISG-11, Rev. 3, also discusses cladding temperature limits and issues guidance on the temperature limits considering hydride reorientation in the vacuum drying process.

NUREG–1567 (NRC, 2000a) recommends that Zircaloy fuel cladding temperature be maintained below 570 °C [1,058 °F] as a suitable criterion for fuel assembly transfer operations. Note that this limit is decreased for high burnup fuels (e.g., greater than 45 GWd/MTU) because of increased internal rod pressure from fission gas buildup (NRC, 2000a). Limited data available on high burnup fuel indicate that increased cladding oxidation, hoop stress, and changes in fuel pellet integrity could lead to cladding failure and possible dispersion of the fuel during handling operations (NRC, 2000a). For fuel burnups exceeding 45 GWd/MTU, it must be shown that the cladding is protected from degradation, which could lead to gross rupture [10 CFR 72.122 (h)(1)], and that the fuel is able to be readily retrieved as required in 10 CFR 72.122(i). NUREG–1567 (NRC, 2000a) states that if the above two requirements cannot be met, the fuel could be enclosed by baskets (to provide further confinement), so degraded fuel would not pose any transportation or retrieval problems.

Because vendors have proposed different configurations of dry storage (i.e., either vertical or horizontal), the effect of orientation on cladding performance, if any, may need to be determined. Johnson and Gilbert (1983) believed that there did not appear to be any theoretical basis for the orientation to affect cladding, because fuel integrity is believed to be principally determined by storage temperature, rod fill gas pressure, and the condition of the cladding. One form of data used to evaluate the horizontal storage mode at that time was the oversea shipment of light water reactor fuel. During these shipments, the fuel was placed in a horizontal orientation for 2 to 3 months at cladding temperatures estimated to be up to 385 °C [725 °F]. These fuel assemblies did not show any evidence of significant damage (Johnson and Gilbert, 1983). Although storage in a horizontal orientation subjects the cladding to different mechanical loading conditions, as compared to a vertical orientation, fuel spacers and storage walls should restrain the rods and prevent any significant fuel rod distortion. In addition, while the temperature profiles will vary somewhat between vertical and horizontal orientations, as long as the horizontal system meets the recommended storage temperature guidelines, cladding behavior would not be expected to differ from that in a vertical orientation (Johnson and Gilbert, 1983).

ISG–1 (NRC, 2007) defines what is considered damaged fuel and provides guidance for the proper classification of spent fuel for transportation and storage. The spent fuel must satisfy the criteria for intact cladding and structural integrity for it to be considered intact. ISG–1 (NRC, 2007) has recognized that for high burnup spent fuel, what is considered intact for storage conditions may not have sufficient cladding integrity to withstand the more severe stresses arising from hypothetical accident conditions that are part of transportation. If, upon examination, the spent fuel contains cladding damage greater than pinhole leaks or hairline cracks, the fuel is considered to be damaged.

Potential mobile (dispersed) radioactive particles inside the dry storage canister can be produced by three sources: (i) disengaged corrosion products; (ii) radioactive gas expelled from a cladding defect; and (iii) fuel particles and fission products from defect geometries (Johnson and Gilbert, 1983). These radioactive gases would be contained in the sealed canister; however, these gases will only be of significance if the fuel needs to be retrieved at some point (i.e., when the canister is opened). Therefore, proper retrieval methods must be defined in advance (Johnson and Gilbert, 1983). NUREG–1536 (NRC, 1997) does indicate that contingencies should be made for particulate materials presented to the system from disengaged corrosion products; however, these particulates will mainly be significant in wet unloading of both boiling water reactor and pressurized water reactor fuel. The particulates can disperse into the pool water and atmosphere during the flooding operations and thereby create possible airborne exposure hazards. This scenario may need to be considered for the case of a

TAD canister containing boiling water reactor fuel because the TAD canister is a sealed canister that must also be capable of being opened at some point if necessary.

During storage in an inert gas environment, creep and/or creep rupture is the dominant deformation mechanism. By its nature, creep is time and temperature dependent. The fuel rod pressure and the resulting circumferential (hoop) stress in the presence of high temperature may produce creep deformation of the fuel rod cladding over time. The internal pressure in the fuel rod is due to the fill gas and the release of fission gas during irradiation.

NUREG-1567 (NRC, 2000a) discusses using accelerated creep tests. However, these tests may result in different creep and/or deformation mechanisms that may occur over different temperature and stress states; this needs to be considered when using this data to evaluate the effect of creep on the fuel cladding. Because creep is a stress-controlled process, the accuracy of the calculated cladding hoop stress is important. The calculation of the hoop stress should take into account (i) reduction in cladding thickness due to oxidation; (ii) the initial fuel rod backfill gas pressure (iii) the additional gas pressure buildup due to fission; (iv) the generation of other gases, such as helium, due to effects caused by the irradiation of any internal cladding coatings; and (v) the thickness of the hydride rim on the outside radius of the cladding (NRC, 2000a).

Note, however, that there are limits on the allowable hoop stress due to the requirement that they be within a certain range to prevent the formation of radial hydrides. Formation of radial hydrides is a function of the hydrogen in the cladding, cladding hoop stress and temperature, cooling rate, material characteristics, and thermal cycling, among others, with hoop stress being the most significant. As given in ISG-11 (NRC, 2003a), data has shown that for low burnup fuel, the cladding hoop stress should be limited to approximately 90 MPa [13 ksi] for un-irradiated fuel and a slightly higher value of 120 MPa [17.4 ksi] for irradiated fuel. As a result, for storage and short-term fuel handling operations (e.g., drying and backfilling), a temperature limit of 400 °C [752 °F] is specified to limit cladding hoop stresses. Other temperature limits for short-term loading operations may be used if it can be shown that the estimated hoop stresses remain below 90 MPa [13 ksi] for the desired temperature.

ISG-11 also discusses thermal cycling as an important consideration with respect to radial hydride formation. Data has shown that the formation of radial hydrides can be limited by restricting the thermal cycling to a cladding temperature difference is greater than 65 °C [149 °F], and by keeping the number of cycles of 10. For a temperature difference less than 65 °C [149 °F], there is no limit on the number of cycles. Thus, thermal cycling limitations will affect the procedures for drying and inerting operations.

Currently, there is no consensus on whether there is sufficient material property data on the cladding behavior of high burnup fuel (typically greater than 45 GWd/MTU for a pressurized water reactor). Therefore, at present, the transportation of high burnup fuel will be handled on a case-by-case basis. The applicant must establish that there is sufficient data to show that the fuel remains subcritical after a hypothetical accident.

The effects of irradiation on the spent fuel cladding are important because the cladding may serve as the initial barrier to prevent the release of radioactive material to the environment and also maintains the configuration of the fuel. In addition, in a feasibility study credit has been taken for the cladding in thermal, shielding, and criticality analyses (Electric Power Research Institute, 2003), and if credit is to be taken, then the integrity of the cladding must be preserved during storage and transportation. For example, the mechanical properties of the cladding that are affected are creep, ductility, and fracture toughness. These properties are important in



maintaining the fuel configuration during dry storage. In addition, for higher burnups, the amount of oxidation, and hydride formation of the cladding increases as well do internal fuel rod pressures, which lead to higher circumferential stresses in the cladding.

Recent data have been obtained from a cooperative program between the DOE and the Electric Power Research Institute. A number of casks were subjected to several performance tests between 1982 and 1992. The primary objective of the pressurized water reactor spent fuel storage cask testing was to obtain data on heat transfer and shielding in addition to limited fuel rod integrity assessments to support at-reactor licensing (Electric Power Research Institute, 2003). Upon completion of these tests, a CASTOR V/21 cask was left loaded with 21 pressurized water reactor fuel assemblies for 14.2 years. Note that the fuel assembly to be examined only had a burnup level of 35.7 GWd/MTU. The temperature history had a peak temperature of 415 °C [779 °F] during thermal testing (simulating the vacuum drying procedure) and a temperature of 350 °C [662 °F] at the beginning of the 14-year storage period (Electric Power Research Institute, 2003). Results indicated that the storage cask and cask internals showed sound structural and seal integrity and that over the 14-year period, there was no detectable degradation in the fuel cladding in terms of rod creep or bow, nor was there any detectable release of fission gas.

Currently, a larger database exists for spent fuel less than the 45 GWd/MTU; however, there is a limited but growing database on burnup values exceeding the 45 GWd/MTU limit. Higher burnup spent fuel will normally result in increased cladding oxidation and hydriding; increased fission gas release results in an increased internal fuel rod pressure, which translates into higher hoop (circumferential) stresses. This is especially true for high-duty fuel cycles (Electric Power Research Institute, 2003). A combination of high temperature and increased hoop stress can lead to deformation and possible rupture of the cladding, a change in the morphology of the hydrides at cooling, and a degradation of the cladding integrity (Electric Power Research Institute, 2003). The most relevant cladding mechanical properties affected include creep, its ductility when subjected to impact due to hypothetical accident conditions, and its fracture toughness. Uncertainty still exists on how the high burnup fuel properties affect the cladding integrity during dry storage and how the cladding properties then impact the transportation, handling, and disposal at the repository (Electric Power Research Institute, 2003).

Finally, NUREG-1567 (NRC, 2000a) [also refer to ISG-5 (NRC, 1999a)] addresses the task of confinement monitoring for redundant sealing when an inert atmosphere is required inside. The general approach has been to pressurize the region between the redundant seals with the same inert gas. Any decrease in the pressure between the seals indicates that the inert gas is either leaking to the outside atmosphere or is leaking into the canister. Note that because the same inert gas is used as in the canister, in-leakage to the canister will not cause any contamination. Also, because the pressure between the seals is greater than the canister, a faulty seal will not result in any radiological consequence (NRC, 2000a). NUREG-1567 (NRC, 2000a) also addresses confinement monitoring with respect to dry storage casks. This requirement applies mainly to mechanical closure seals, such as bolted closures. For welded closures, it has been accepted that such seal monitoring is not required. This approach is consistent with the fact that the other welded joints used in fabrication are not subject to monitoring. However, NUREG-1567 states that because a continuous monitoring system is not necessarily used, periodic evaluation should provide timely and corrective measures to maintain safe storage conditions if closure degradation occurs (NRC, 2000a). For storage casks with closure lids that are tested to be leaktight (ANSI N14.5), such monitoring may be unnecessary. This may be important for the TAD canister because periodic monitoring will not be possible during disposal.

## 1.4 Title 10 Code of Federal Regulations: Part 63

The TAD canister will be required to satisfy 10 CFR Part 63 for aging and disposal at the potential Yucca Mountain repository. NRC will carry out a risk-informed, performance-based regulatory program using concepts of reasonable expectation (10 CFR 63.304) and reasonable assurance.

With respect to canister sealing, ISG-18 (NRC, 2003b) addresses, as discussed in the previous section, the final closure welds of austenitic stainless steel canisters. ISG-18 (NRC, 2003b) states that failure of welds that are made in accordance with ISG-15 (NRC, 2001) is rare and is usually caused by fabrication errors. These types of flaws are immediately detectable by postweld examination techniques (NRC, 2003b). The guidance for the weld design and examination techniques to be used is given in ISG-15 (NRC, 2001), and staff believe that any weld flaws that could lead to failure would be detected using these techniques. Therefore, the guidance in ISG-18 (NRC, 2003b) concluded that “satisfying both ISG-15 and ISG-18 provides reasonable assurance that no credible leakage can occur from the final closure welds” (NRC, 2003b). Therefore, the inert environment is expected to be maintained and no cladding degradation will occur. As given in NUREG-1536 (NRC, 1997) using the specified vacuum drying method, a limit on the maximum quantity of oxidizing gases to a concentration of 0.25 volume percent should reduce the amount of oxidants to levels below which cladding degradation can be expected. With no credible leakage, the low oxidation, inert environment should be preserved. ISG-18 also notes that the qualification standards in this guidance provide a sufficient alternative to the fabrication, periodic, and preshipment leak-testing requirements (ANSI N14.5) for the final closure welds (NRC, 2003b).

The concept of importance to waste isolation (as discussed in 10 CFR 63.102) refers to those engineered and natural barriers whose function is to provide a reasonable expectation that high-level waste can be disposed of without exceeding the requirements of 10 CFR 63.113(b) and (c). Of particular concern here are the engineered barriers (i.e., the drip shield, waste package, and the spent fuel rod cladding). Specifically, 10 CFR Part 63 requires a multiple barrier approach for the repository. The requirement for multiple barriers ensures that repository performance is not entirely dependent on a single barrier, thereby making the total system more tolerant of failures.

The integrity of the fuel cladding must continue to be assessed. Previously, when using the concept of dual-purpose canisters for transportation and storage, additional fuel handling was to be performed at the repository in which the fuel would be repackaged into the waste package. This repackaging would have provided an opportunity to determine the current condition of the fuel. Now, however, with the use of a sealed TAD canister, knowing the exact condition of the fuel cladding before aging and disposal will not generally or routinely be possible.

As stated in 10 CFR 63.142(a), 10 CFR 63.21(c)(20) requires DOE to describe in a safety analysis report a quality assurance program to be applied to all important to safety structures, systems, and components and to the natural and engineered barriers that are important to waste isolation. A quality assurance program enhances confidence in the design and characterization of barriers that are important to waste isolation—one of which is potentially the fuel rod cladding. 10 CFR Part 63 requires that DOE implement quality assurance requirements that are commensurate with the significance of the internal materials and components (e.g., cladding). In 10 CFR 63.142(n), for particular products, DOE is directed to provide

protective environments, such as an inert gas atmosphere, moisture content levels, and temperature levels, all of which are important to cladding integrity.

By observing the requirements set forth in 10 CFR Parts 71 and 72, the integrity of the low burnup fuel rod cladding is preserved by imposing thermal limits during drying and inerting operations and during storage. DOE reports that low burnup fuel data collected from industry indicate that handling and transportation do not result in significant increases in cladding degradation. In addition, based upon data collected from multiple utility sources, the amount of initial out-of-reactor cladding damage is also low (DOE, 2007b).

However, DOE takes the position that because the TAD canisters are sealed at the loading facility, inspection of the cladding integrity after transportation and arrival at the repository may not be possible. Currently, DOE does not take credit for the spent fuel cladding as a barrier to flow and transport of radionuclides (DOE, 2007b). DOE presents the following two arguments for this position. First, DOE has stated that because of “difficult-to-quantify uncertainties” (DOE, 2007b) pertaining to the as-received condition of the cladding upon arrival of the TAD at the repository, no credit was to be taken for the cladding capabilities in the Total System Performance Assessment model. This position may be based upon the current lack of data regarding the loading, shipment, and handling for moderate to high burnup commercial spent nuclear fuels. By not taking credit for the cladding, DOE would decrease the possible risk associated with not inspecting the spent fuel sealed inside the TAD. Second, based on DOE’s current Total System Performance Assessment model, cladding performance was not expected to be a major factor in reducing dose estimates (DOE, 2007b). Considering that the Total System Performance Assessment compliance model for the license application is to be predicated on the concept of reasonable expectation, the extent to which this assumption is conservative will need to be evaluated.

## 2 SURVEY OF TOPICS RELATED TO DRYING AND SEALING

The following sections detail the process and methodology of sealing and drying of typical spent fuel canisters. Because the proposed TAD canister is expected to be based on current multipurpose canister designs, the following discussion will focus on the drying and sealing operations of multipurpose canister systems taken from currently available documentation (e.g., safety analysis reports).

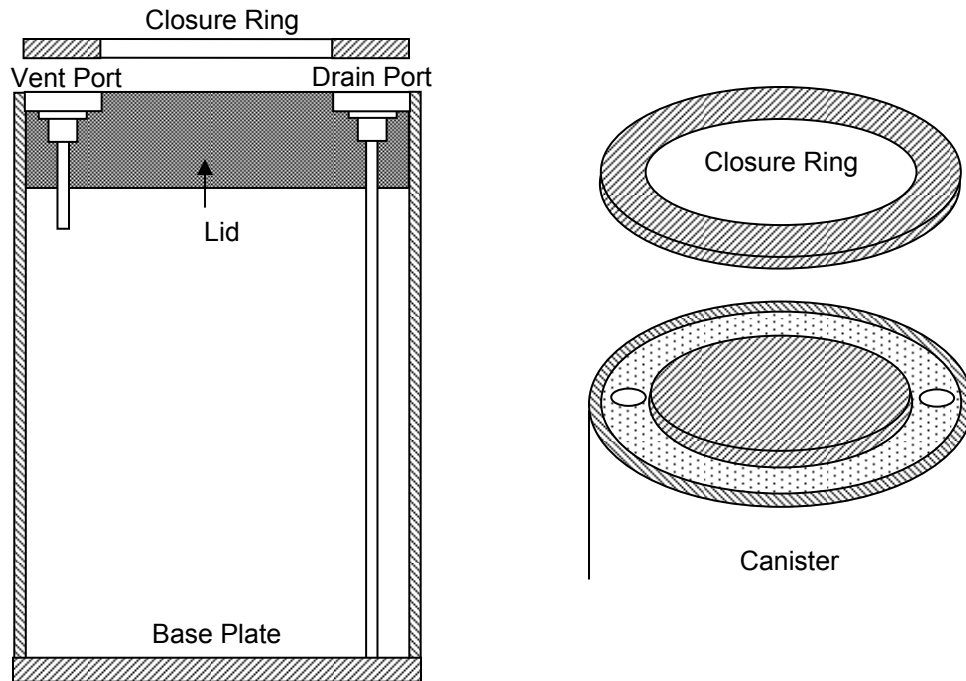
For normal storage conditions, a typical spent fuel, multipurpose canister uses multiple confinement barriers provided by the fuel cladding and the canister enclosure vessel to assure that there is no release of radioactive material to the environment. Weld examinations, including multiple surface and volumetric examinations, hydrostatic testing, and leakage rate testing on the canister lid weld and the vent and drain port cover plate welds, assure the integrity of the canister closure. The multipurpose canister is backfilled with an inert gas (typically helium) to protect against cladding degradation. The presence of helium inside of the canister also serves as the tracer for the helium leak test to be performed in accordance with ANSI N14.5.

### 2.1 Sealing

The complete confinement boundary of a typical spent nuclear fuel canister consists of the canister shell, a bottom baseplate, a lid that may include vent and drain ports, a closure lid, and the associated welds of each of these components. All of these components would form a totally seal-welded vessel for the storage of the spent fuel assemblies.

One method of sealing the canister that Doman (2004) proposed utilizes a shield plug and an outer seal plate. Figure 2-1 shows a general schematic of a canister seal. The shield plug, shear ring, and outer seal plate (ring) are separate components and would be installed after the spent fuel is loaded. The shear ring is welded to the shield plug and to the canister body, and the canister is filled with helium. Once this has been accomplished, the outer seal plate is welded to the canister. The inner seal weld is tested for leaktightness by measuring the amount of fill gas present between the shield plug and the outer seal plate. Subsequently, a fill gas is supplied to the space between the outer seal plate and the shield plug. Samples are taken from this space to determine the leaktightness of the outer seal plate weld. Thus, the outer lid welded to the upper part of the canister shell provides redundant sealing of the confinement boundary.

An example of the shield plug design is used in the approach of the Holtec HISTORM 100 multipurpose canister system (Holtec International, 2006). The multipurpose canister consists of a shell and a baseplate welded at the bottom. Fabrication and welding of these components are done prior to loading with the spent fuel. After being loaded with spent fuel, the multipurpose canister lid (previously referred to as a shield plug) is welded to the multipurpose canister shell. Cover plates used to seal the vent and drain ports are welded to the canister lid after the draining, moisture removal (e.g., vacuum drying), and the subsequent helium backfill. A closure ring welded to the multipurpose canister lid covers the vent and drain port cover plate welds and provides a redundant closure of the canister. Note that the closure ring is similar to the outer seal plate (ring) as Doman (2004) describes.



**Figure 2-1. General Representation of Canister and Seal Components**

Note that other than the vent and drain ports, no other penetrations into the confinement barrier of the multipurpose canister exist. Other manufacturer [e.g., FuelSolutions W21 Canister (EnergySolutions Spent Fuel Division, 2006), NAC-MPC (NAC International, 2004)] canisters are also of the seal-welded type. Common to these designs are the use of vent and drain ports as the only penetrations into the confinement boundary. In addition, redundant sealing is used that consists of the shield plug, inner closure plate, outer closure plates, and the associated welds.

The complete closure of a canister involves the following operations:

- Welding and nondestructive testing of the shield plug
- Draining, drying, and inerting the canister interior
- Leak testing
- Welding and nondestructive testing of the vent and siphon port covers
- Welding and nondestructive testing of the final closure lid on the canister

The closure welds typically follow the American Society of Mechanical Engineers Boiler and Pressure Vessel Code standards. For example, in the safety analysis report of the Holtec Hi-Storm System, Holtec states that using procedures in full compliance with Section III of the ASME code ensures high quality welds (Holtec International, 2006). The guidance of ISG-15 (NRC, 2001) recommends using the American Society of Mechanical Engineers welding code. NUREG-1567 (NRC, 2000a), which provides guidance for the closure of dry storage canisters, recommends that the closure welds be leak tested using helium. NUREG-1567 also specifies that hydrostatic or other pressure tests are not required if structural analysis has determined a

minimum factor of safety of 1.5 against design pressure. ISG-15 (NRC, 2001) specifies that the closure welds may be either full thickness penetration or a partial penetration groove weld. For a partial penetration weld, the minimum depth of the groove should be equal to or larger than the canister shell thickness (NRC, 2000a). Also, for partial penetration groove welds, the maximum clearance between the canister lid and wall should be sufficiently small to allow a proper weld and cannot exceed the clearance allowed in the weld procedure specifications. As per ISG-15, the minimum tensile strength of the weld metal must be equal to or larger than the base metal to prevent weld metal failure. The strength of the weld is based on the nominal weld area and the stress intensity values for the weaker of the two materials being joined.

A TAD canister closure study (Bechtel SAIC Company, LLC, 2007) examined a number of welding processes that could be used for the TAD. The results of this study recommended that a gas tungsten arc welding method be used. It is the Bechtel SAIC Company, LLC (2007) position that gas tungsten arc welding produces a consistent, high quality weld that meets the American Society of Mechanical Engineers Boiler and Pressure Vessel code (Bechtel SAIC Company, LLC, 2007). Gas tungsten arc welding can produce clean, high quality welds with a minimum of defects and is typically used in the nuclear industry where strict conformance to code is required (Bechtel SAIC Company, LLC, 2007). One of the disadvantages of gas tungsten arc welding is that it is slower than other welding processes such as gas metal arc welding. The main advantage of the gas metal arc welding process is that it has high deposition rates making it a fast process (Bechtel SAIC Company, LLC, 2007). During the process of welding metal together, a scale or slag develops over the surface of the weld; gas metal arc welding produces little or no slag that would normally need to be removed before a subsequent pass is made (Bechtel SAIC Company, LLC, 2007). Thus one possibility was presented in which gas tungsten arc welding could be used for the root pass welds and gas metal arc welding used for the additional weld passes. However, the speed of the gas tungsten arc welding process can be increased by using preheated feed wire to increase weld metal deposition (Bechtel SAIC Company, LLC, 2007). However, the final recommendation for using gas tungsten arc welding is based on the fact that it produces the highest quality weld and can be automated (Bechtel SAIC Company, LLC, 2007).

With regard to nondestructive testing methods, NUREG-1567 specifies that for austenitic stainless steels (e.g., 304, 316, and 316L), the closure welds may be examined by either an ultrasonic method or the liquid penetrant type of examination. If a liquid penetrant method is used, the examination is to be made progressively on the root layer (the lesser of one-half the weld joint thickness and at ½-inch intervals thereafter and at the final surface). As given in ISG-4 (NRC, 1999b), the NRC does not regulate to the standard of absolute assurance, only to the standard of adequate protection. While ultrasonic method examination is preferred because it is a volumetric method, liquid penetrant examination is considered to provide reasonable assurance that, specifically for austenitic stainless steels, the possible critical flaws will be identified. Ultrasonic method examination is subject to false indications of a flaw, and considerable operator skill with performing the test and interpreting the results may be necessary. A false indication may signify that a flaw in the weld has been detected, which may erroneously be considered to violate the minimum flaw size criteria. If not correctly interpreted, these false indications could lead to unnecessary grinding out of the weld material (NRC, 1999b).

As part of the TAD closure study (Bechtel SAIC Company, LLC, 2007) mentioned previously, alternative nondestructive examination methods were also recommended. Bechtel SAIC Company, LLC (2007) recommends that the liquid penetrant method is best suited for detecting surface flaws (Bechtel SAIC Company, LLC, 2007). Some of the disadvantages of the liquid

penetrant method are: (i) the inspector must have direct access to the surface being tested, (ii) precleaning is critical because contaminants can mask defects, and (iii) multiple process operations (i.e., progressive examination of each weld pass) must be performed and controlled. Bechtel SAIC Company, LLC (2007) does recommend using a combination of visual, eddy current, and ultrasonic methods. One advantage of these methods is that they can be performed remotely without any hands-on procedures. Note that because the visual and eddy current methods only detect surface flaws, the ultrasonic method would be used to examine the flaws deep within the weld. These recommended methods (i.e., visual, eddy current, and ultrasonic) are standard methods used in the nuclear power industry (Bechtel SAIC Company, LLC, 2007).

## **2.2           Drying Processes**

With respect to drying, three questions need to be answered:

- How dry does the canister need to be?
- What procedures and processes will be used for drying?
- How will it be determined that the canister is dry?

Recently, the American Society for Testing and Materials (ASTM) has issued a standard guide, C 1553 (ASTM, 2008), to address some of these issues.

One issue is determining how dry the spent nuclear fuel must be to prevent problems such as corrosion, canister pressurization, and fuel retrievability issues. The presence of sludge, crud, and other hydrated materials poses additional difficulties because of the ability of these materials to hold water and to resist drying. Water may be retained in spent nuclear fuel cladding having hairline cracks or holes and in the capillaries of Boral plates if used. In place of Boral plates, the use of metal matrix composites have been reported to reduce the amount of bound water (Washington Nuclear Corporation, 2005). Therefore, the drying method used must successfully remove bound water, which is only accomplished when sufficient energy is applied to break the bonds. However, as discussed in Section 1 of this report, the drying process must not damage the fuel, particularly the cladding. For example, thermal cycling could cause hydride reorientation in the cladding.

Selecting the appropriate drying procedures and processes should be based upon a number of factors [(e.g., the evaluation of the degree of damage to the spent fuel, the form of water in the canister, degree of self-heating that may contribute to the drying process, and interaction of the water with the spent fuel and canister components (ASTM, 2008)]. The drying temperatures, the time at a given vacuum level, and the number of backfill and reevacuation cycles also are parameters that should be addressed in the drying process specifications (ASTM, 2008). Decay heat by itself is usually insufficient to prevent the formation of ice during drying. As will be discussed, stepped evacuation methods with or without helium backfill have been used in commercial drying processes to prevent the formation of ice.

As mentioned previously, a key issue is determining adequate dryness of the canister. Three methods have been proposed to determine dryness: (i) pressure measurements, (ii) internal hydrogen concentration, and (iii) process knowledge (ASTM, 2008). A pressure rebound test is the current method used. Specifically, for commercial spent nuclear fuel, 3 torr must be maintained for 30 minutes to indicate that less than one mole of residual gas from trapped water or ice remains in the canister (ASTM, 2008). Measuring the internal gas composition to

determine the hydrogen concentration could indicate the amount of water that was present and released as a result of radiolysis (ASTM, 2008). This approach may not be practical for the TAD, because it will be completely sealed at the utility site. Finally, process knowledge involves using the spent nuclear fuel drying and storage history to verify that no significant amount of water could be present within the canister. This approach, however, requires accurate records of the spent fuel history's irradiation, drying, and storage and may not be practical for other than pristine spent fuel (ASTM, 2008).

The main objective of the drying process is to remove any moisture entrapped inside the canister and the fuel cladding. Also, as the canister temperature increases due to the spent fuel, any water that remains in the canister would boil and turn to steam, which will cause a significant pressure increase (DOE, 2003). During drying, the process must also ensure that the temperature remains at a level such that no high thermal strains, which could rupture the fuel cladding, are generated. There are two primary means of canister drying: a cold vacuum drying system, or a hot gas drying system (DOE, 2003).

Vacuum drying is considered to be the industry standard (DOE, 2003). This method of drying is referred to as "cold" vacuum drying because external heat is not normally applied to the drying vessel. Note, however, that there is spent nuclear fuel decay heat that helps evaporate the water and assists in drying. In this method, the canister is drained of the bulk of the liquid water and a vacuum system is connected to the canister to create a subatmospheric state within the canister. The reduction in pressure within the canister reduces the boiling temperature of the water thereby vaporizing the water contained within the canister as well as the moisture present on the spent fuel assemblies. One benefit of the vacuum system is that air or other gases in the canister are also removed thus making subsequent inerting of the canister simpler.

As given by NUREG-1536 (NRC, 1997), a required vapor pressure of 3 torr (1 torr = 1 mm Hg) must be attained. This pressure must be held constant for 30 minutes with the vacuum pump isolated to verify that the amount of moisture removal is adequate. As specified in NUREG-1536, provisions should be made to prevent formation of ice in the evacuation system. Icing can occur due to the cooling effects of system depressurization during evacuation as well as effects of the water vaporization itself. Therefore, a staged or stepped evacuation process (e.g., adjustment of the flow rate) is used to eliminate the possible formation of ice inside the canister and vacuum lines. With regard to ice formation in the canister, NUREG-1536 states that it is less likely to occur due to the decay heat from the spent fuel. However, icing could likely occur in breached fuel around the small opening during rapid pump rundown.

As mentioned previously, the internal pressure is reduced and held for a period of time to ensure that all moisture (liquid water) has evaporated. As a result, there may be some degassing of the spent fuel assemblies due to water contained within the cladding. The surface condition of the fuel rod (e.g., the presence of crud) will also contribute to the amount of water to be removed from the surface of the fuel. The crud itself may spall off the rods due to handling, and small cracks that could contain water may be present. The cooling effects due to the depressurization may also be important with respect to the integrity of the fuel cladding. NUREG-1536 discusses whether excessive cooling rates could produce thermal stresses that may cause fuel cladding and fuel rod component damage. Thus, possible formation of ice in the canister and thermal stresses in the cladding are important points to consider in the vacuum drying process. Although specific details are not given in the TAD proof-of-concept reports, EnergySolutions (2007), NAC International (2007), and Transnuclear Inc. (2007) discuss the use of the vacuum drying process in their proposed TAD.



An alternative to the vacuum drying process is the forced helium dehydration also generally referred to as a forced gas flow process (Singh, 2007). A hot gas system removes the moisture from the canister and fuel assemblies by using the difference in humidity of the hot gas sent into the canister, which is at low humidity, and the gas surrounding the fuel assemblies, which has a higher humidity. The hot gas entering the canister must be dried prior to injection to maximize the amount of water that can be absorbed. The drying utilizes either a once-through or a recirculating process. The once-through system is considered inefficient because the vaporization process saturates regardless of the amount of gas pumped into the canister.

In the forced helium dehydration process, an inert gas, such as helium, is circulated through the canister. The helium gas leaves the canister at approximately 177 °C [350 °F] and is first cooled by a condenser to remove some of the water vapor. The gas is further cooled by freeze-drying {to approximately -6 °C [21 °F]} to remove additional moisture. Monitoring the cooling temperature of the gas assures that the gas is dry. The gas is then superheated {at approximately 149 °C [300 °F]} in an auxiliary heater and passed to the canister. Because the spent nuclear fuel rods also radiate heat, the hot dry helium gas is further heated to 177 °C [350 °F]. The superheated helium then absorbs the water vapor from the cask cavity. As long as the gas flows, the canister internal vapor pressure will continue to drop as well as the vapor pressure of the exiting wet helium gas. The canister vapor pressure will continue to drop until it equilibrates with the vapor pressure of the heated dry gas entering the canister. Controlling the temperature of the helium exiting the condenser ensures that the vapor pressure is low enough to reach a target pressure of 3 torr or lower, which controls the degree of dryness in the canister. As long as there is a sufficient amount of dry helium gas flowing into the canister, the vapor pressure can be maintained.

The dry gas system may have difficulties with pool water that has collected at the bottom of the canister because water is only removed when gas flows over water surface (DOE, 2003). Holtec has estimated that between 16 and 38 hours will be required to dry a proposed TAD canister (Bencel, 2007). This predicted time is based upon Holtec's calculations considering that their current multipurpose canister system is designed for existing dry-storage dual and multipurpose canisters. Because the volume of the TAD canister is smaller than the dual-purpose canisters and multipurpose canisters, these calculations should be applicable to the TAD canister (Bencel, 2007).

Table 2-1 compares the standard cold vacuum and forced helium (gas) dehydration methods as discussed previously.

## **2.3 Leak Testing**

The welded construction of the spent fuel canister along with the welded shield lid, port covers, and closure ring should ensure that containment is maintained during storage and transportation. Because the canister relies on welds for its redundant closure, the spent fuel canister's confinement capability needs to be verified using pressure (i.e., hydrostatic or pneumatic) testing, nondestructive testing of the welds, and helium leak testing. It is important that the amount of helium is preserved in the canister to maintain an inert atmosphere, which in turn prevents cladding degradation over the complete storage period of the cask. In addition, by definition in ANSI N14.5 (American National Standards Institute, 1998), a package (e.g., spent fuel canister) is considered leaktight if the degree of containment precludes any radiologically significant release of radioactive materials. ANSI N14.5 states that this degree of containment is achieved by demonstration of a leakage rate less than or equal to  $1 \times 10^{-7}$  ref-cm<sup>3</sup>/s

| <b>Table 2-1. Comparison of Vacuum Drying and Forced Helium Dehydration Methods</b>   |  |  |
|---|--|--|
| <b>Drying Method</b>  | <b>Vacuum Drying</b>   | <b>Forced Helium Dehydration</b>   |
| Criterion   | Maintain vacuum of 3 torr for 30 minutes   | Exit recirculating gas temperature of $-6\text{ }^{\circ}\text{C}$ [ $\leq 21\text{ }^{\circ}\text{F}$ ] for $\geq 30$ minutes*  |
| Efficiency/Process Time   | Estimated time range of 30 to 65 hours   | Estimated to be 16 to 38 hours†  |
| Advantage   | <ul style="list-style-type: none"> <li>- Industry standard and accepted method of drying‡</li> <li>- Simple air-cooled vacuum drying systems eliminate need for external heating and cooling units</li> <li>- Less expensive to purchase and maintain</li> </ul>   | <ul style="list-style-type: none"> <li>- Eliminates the possibility of ice formation in canister</li> <li>- Maintain cladding temperature due to heat transfer of recirculating gas</li> </ul>   |
| Disadvantage  | <ul style="list-style-type: none"> <li>- Uncertainty of possible formation of ice in canister</li> <li>- Difficult to judge whether there has been sufficient time for sublimation of ice</li> <li>- Note: At pressures below 4.7 torr, ice may form in canister requiring 3 to 4 times longer for removal§</li> </ul> | <ul style="list-style-type: none"> <li>- Internal configuration (e.g., basket structure) may prevent adequate flow or circulation of gas over all surfaces in canister</li> <li>- May have difficulty with pooled water in canister bottom</li> <li>- Requires more complicated equipment setup (e.g., helium circulator), preheat and chiller module</li> </ul> |
| <b>References</b><br>*Holtec International. "Final Safety Analysis Report for the Hi-Storm 100 Cask System." Rev. 4. USNRC Docket No. 72-1014. West Marlton, New Jersey: Holtec International. April 4, 2006.<br>†Bencel, K. "TAD/STC Drying and Inerting Calculation." 000-M0C-MR00-00100-000-00A. Washington, DC: Office of Civilian Radioactive Waste Management. 2007.<br>‡NRC. NUREG-1536, "Summary Review Plan for Dry Cask Storage Systems." Washington, DC: U.S. Nuclear Regulatory Commission. 1997.<br>§Kelly, J.G. "The Myths and Realities of Vacuum Drying." Dry Storage Information Forum, May 15-17, 2007. Clearwater Beach, Florida: Published on CD ROM. Washington, DC: Nuclear Energy Institute. 2007.<br>Note: [1 torr = 1 mm Hg]<br>[1 °F = 1.8 °C + 32] |  |  |

[ $6.1 \times 10^{-9}$  ref in<sup>3</sup>/s] of air at an upstream pressure of 1 atmosphere absolute [14.7 psi] and a downstream pressure of 0.01 atmosphere absolute or less.

As an example, in the Holtec HI-STORM 100 cask system (Holtec International, 2006), the confinement boundary of the multipurpose canister is defined by the multipurpose canister shell, baseplate, multipurpose canister lid, port cover plates, closure ring, and associated welds; thus, the multipurpose canister is a completely seal-welded vessel. Holtec states that leak testing is to be performed on the welds of the vent and drain port covers and the closure ring of the multipurpose canister lid to verify weld integrity. As specified by Holtec, the helium leakage test of the vent and drain port cover plate welds shall be performed using a helium mass

spectrometer leak detector. A preliminary review of the safety analysis reports for the NAC International Multipurpose Cask (NAC International, 2004) also specifies that the leak detector to be used is of the helium mass spectrometer type (American National Standards Institute, 1998).

There are a number of available leak test procedures; however, the selection of an appropriate procedure depends on the required measurement sensitivity and its applicability to the type of package to be tested. The mass spectrometer leak detector is used in conjunction with a sniffing type of collection method that utilizes helium as the tracer gas. This method involves pressurizing the canister with helium, which has been done in the inerting operation, and moving the mass spectrometer leak detector probe across the weld areas to be tested. Based on ANSI 14.5, this form of test has a nominal sensitivity of  $10^{-3}$  to  $10^{-6}$  cm<sup>3</sup>/s [ $6.1 \times 10^{-5}$  to  $6.1 \times 10^{-8}$  in<sup>3</sup>/s]. It is noted that as part of the procedure, the leak detector is checked using a known helium source immediately before and after the test to determine whether there is any unknown leak detector failure.

## **2.4 Summary and Conclusions**

As part of the spent fuel processing, canisters are loaded with the spent fuel under water. After draining the majority of the water, the remaining water in the canister must be removed using vacuum or hot gas drying to prevent corrosion and hydriding of the spent fuel rods as well as corrosion of other internal canister components. The most significant issues to be considered in the selection of the drying process are. (i) maintaining an acceptable cladding temperature during the drying period, (ii) selecting a drying period that is not excessively long, and (iii) preventing the possible formation of ice. The two drying processes in use today are vacuum and hot gas drying. The vacuum drying process involves lowering the cover gas system pressure below the vapor pressure of the water. It is the traditional practice to consider the canister to be dry when the system pressure, 3 torr, remains constant for a specific time period (30 minutes). However, large amounts of residual water could require longer drying times. Therefore, the vacuum cycle must be slow enough to allow full removal of all the water. There may also be the possibility of the freezing of residual water due to evaporative cooling. In the forced hot gas drying process, the fuel assembly is heated by the hot gas to evaporate the water. Helium should be used for the hot gas to preserve the inert environment to prevent cladding oxidation. The time required to dry the canister depends upon the heat input and flow rate of the hot gas. The canister is considered dry when the moisture content of the exiting gas equals the moisture content of the entering gas. Review of a number of TAD proof-of-concept reports released by Transnuclear, Inc. (2007), NAC International (2007), and EnergySolutions (2007) yielded no new details on the TAD drying processes. Holtec proposes to use forced gas dehydration, while the remaining vendors have indicated that vacuum drying may be used. Sealing of the canister is accomplished by welding. This report focused on two different types of welding processes: gas metal arc welding and gas tungsten arc welding. Gas metal arc welding has the main advantage that it has high deposition rates making it a fast process. However, the gas tungsten arc welding method produces a high quality weld with a minimum of defects and is typically used in the nuclear industry where strict conformance to code is required. Note that the deposition rate of the gas tungsten arc welding process can also be increased by using a preheated fee wire. Standard helium leak testing, satisfying the criteria set forth in ANSI 14.5, will be used to verify that the welds are leaktight. Nondestructive testing will also be used to check the welds for flaws. Dye penetrant testing is one of the most commonly used techniques. However, this method is best used to detect surface flaws; otherwise, volumetric methods such as ultrasonic testing are to be used. A TAD canister closure study

Bechtel SAIC Company, LLC (2007) performed recommended that nondestructive testing methods that can be performed remotely be adopted so as to limit worker dose.

In summary, the present review of the currently available documents pertaining to drying and sealing has not uncovered any previously undocumented material. Thus far, this review has not raised any new issues with regard to drying and sealing. The next phase of this project will focus on Final Safety Analysis Reports of the current multipurpose canister (e.g., Holtec International, 2006; NAC International, 2004; EnergySolutions, Inc., 2006) and cask designs. These reports may provide additional details of the drying and sealing processes used.

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## **APPENDIX A**

### **SUPPLEMENTAL DRYING/SEALING GUIDANCE NEEDS FOR TRANSPORTATION, AGING, AND DISPOSAL CANISTER DESIGN**



## **Appendix A**

# **SUPPLEMENTAL DRYING/SEALING GUIDANCE NEEDS FOR TRANSPORTATION, AGING, AND DISPOSAL CANISTER DESIGN**

**Prepared for**

**U.S. Nuclear Regulatory Commission  
Contract NRC-02-07-006**

**Prepared by**

**T. Wilt**

**Center for Nuclear Waste Regulatory Analyses  
San Antonio, Texas**

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## QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT

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# 1 WELDING AND DRYING ISSUES

## 1.1 Introduction

The typical spent fuel canister consists of a shell assembly that is a high integrity pressure vessel that confines the radioactive materials during dry storage and maintains an inert helium environment for the spent nuclear fuel. Another consideration is the potential oxidation of the cladding. The helium atmosphere assures corrosion protection and enhances heat removal. The canister also has an internal basket assembly for the geometric spacing and criticality control for both storage and transport conditions. One of the primary functions of the canister is to maintain the integrity of the spent fuel rod cladding during normal operating conditions. In the context of dry storage, one of the dominant failure mechanisms of the cladding when subjected to high temperature over long periods is thermally induced creep. Thermally induced cladding failure in zircaloy-clad fuel can be controlled by maintaining a peak cladding temperature that is sufficiently low so as to minimize the amount of creep during storage.

Proper drying and inerting of the canister interior are important factors that affect spent fuel integrity. In the following sections, the safety analysis reports of four representative canister systems will be reviewed with regard to their specific drying and sealing (welding) procedures.

## 1.2 Safety Analysis Reports

### 1.2.1 FuelSolutions™ W21 Canister

#### 1.2.1.1 Welding

The possible presence of hydrogen in the area where the welding takes place affects sealing (welding) of the canister (EnergySolutions Spent Fuel Division, Incorporated, 2006a). A first step in the canister sealing process is to weld the inner canister lid to the canister shell. To perform the welding, the water level in the canister is lowered by a set amount to provide clearance. In this void thus created, hydrogen gas may be present; therefore, purging is performed before welding to eliminate the potential for a hydrogen gas burn event. In this safety analysis report, a hydrogen concentration of 0.4 percent (which is defined as 10 percent of the lower explosive limit) by volume is set forth as the safety criterion (EnergySolutions Spent Fuel Division, Incorporated, 2006b). If hydrogen exceeding this concentration limit is detected, then the void space under the inner closure lid is purged with a “welding grade” argon gas. The argon gas supply (introduced through the vent port) is to be maintained until the root pass of the closure weld is completed. Subsequently, as part of the closure process, a dye penetrant inspection of the root pass weld is performed. If the inspection indicates that any weld repairs are necessary, the argon gas purge is reestablished and maintained until the repairs are completed.

Once the inner canister lid has been welded, a compressed inert gas such as argon, helium, or nitrogen is used to force the bulk of the water from the canister through the drain port. The compressed gas is specified to have a maximum pressure of  $2.07 \times 10^5$  Pa [30 psig]. After the water stops flowing from the canister, it is recommended to continue to purge with compressed inert gas for a minimum of 30 minutes (EnergySolutions Spent Fuel Division, Incorporated, 2006b).

### 1.2.1.2 Drying

As part of the vacuum drying process, the pressure inside the canister is gradually reduced in a stepwise manner to eliminate the formation of ice in the vacuum lines. The pressure, as specified in EnergySolutions Spent Fuel Division, Incorporated (2006a), is reduced in the following increments: 100, 50, 25, 15, 5, and 3 torr [1.93, 0.97, 0.48, 0.29, 0.10, and 0.06 psi]. As each internal pressure level is attained, the vacuum process is stopped. At this point, the pressure inside the canister will increase as the water and other volatiles evaporate. Once this new pressure level has stabilized, the vacuum process is resumed and the pressure is lowered to the next level. Once the vacuum pressure of 3 torr [0.06 psi] is reached, the pressure is monitored and must be maintained at this level for a minimum of 30 minutes. This last step satisfies the test criteria as given in Knoll and Gilbert (1987). Note that it may be necessary to repeat some steps of the vacuum process depending on the rate and extent of the pressure increase during the hold time.

Once the vacuum criterion of 30 minutes at 3 torr [0.06 psi] has been satisfied, the canister is pressurized with 99.995 percent pure helium gas to a minimum of  $8.62 \times 10^4$  Pa [12.5 psig] (EnergySolutions Spent Fuel Division, Incorporated, 2006a). At this pressure, a helium leak rate test is performed on the welds of the inner top closure plate and the vent and drain port bodies (EnergySolutions Spent Fuel Division, Incorporated, 2006a). Note that this type of test satisfies both the helium leak rate test as well as pneumatic pressure testing. If the canister satisfies the leak test, the canister should be reevacuated until a stable vacuum pressure of 3 torr [0.06 psi] or less has been achieved and held for a minimum of 30 minutes. The final step is to connect a supply of 99.995 percent pure compressed helium gas and repressurize the canister, allowing helium to flow into the canister cavity until the helium backfill density requirement is met. The FuelSolutions™ W21 canister helium backfill density is in the range of 0.0368 to 395 g-moles/liter [ $2.295 \times 10^{-3}$  to  $2.47 \times 10^{-3}$  lb-moles/ft<sup>3</sup>] (EnergySolutions Spent Fuel Division, Incorporated, 2006b).

The strength and stiffness of the zircaloy depends on its mechanical properties such as elastic modulus and yield strength. High temperature exposure can anneal the fuel cladding material (EnergySolutions Spent Fuel Division, Incorporated, 2006b). This annealing process could reduce the strength of the cladding. The structural performance of the spent fuel rods is also important when considering hypothetical accident conditions such as an end drop of the canister. For this case, the spent fuel rod may fail due to buckling. Therefore, it is important to control the cladding temperatures. With these points in mind, for the FuelSolutions™ W21 canister, the drying time is limited to 12 hours. This time limit is based upon the consideration of maintaining acceptable cladding temperatures. It is noted that if additional vacuum drying time is necessary, a hold period of 4 hours for cooling under the helium gas backfill is required before the beginning of another 8-hour vacuum drying period (EnergySolutions Spent Fuel Division, Incorporated, 2006b). This cycle of cooling under helium backfill and reevacuating should be performed as many times as required to satisfy the canister vacuum pressure limit.

## 1.2.2 Transnuclear: The Standardized NUHOMS® Horizontal Modular Storage System

### 1.2.2.1 Welding

This safety analysis report discusses the dry shielded canister that is used as part of the standardized NUHOMS® Horizontal Modular Storage system. The pressure-retaining containment boundary of the dry-shielded canister consists of a cylindrical shell with top and

bottom cover plate assemblies and shield plugs to limit the worker doses during drying, sealing and handling operations. Redundant seal welds are used for the joints between the shell and the top and bottom cover plates. Note that the bottom seal welds are made during the fabrication of the dry-shielded canister, while the redundant top welds are made after the loading of the spent fuel. As part of the redundant sealing of the canister, the siphon and vent ports utilize cover plates, which are also welded to the canister top assembly after the drying operations are completed. By utilizing the redundant weld closure system, it is assured that no single failure of a weld will result in the breach of the canister (Transnuclear, Incorporated, 2006).

Because the spent fuel cladding is considered to be the first barrier of containment for the spent fuel (Transnuclear, Incorporated, 2006), protection of the cladding is important. The spent fuel cladding may rupture and fail due to its volumetric expansion, which is caused by oxidation. Cladding protection is accomplished by establishing an inert environment, in particular, a helium atmosphere, which prevents the ingress of oxygen into the dry shielded canister (Transnuclear, Incorporated, 2006). Because helium has a small atomic diameter, it has the capability to leak through valves, mechanical seals, and small passages (Transnuclear, Incorporated, 2006). With the careful design of vessel closures, negligible leakage rates are possible. In this regard, to minimize leakage potential, the dry-shielded canister is designed with no mechanical penetrations other than the siphon and vent ports. Note, however, that these ports are also later covered with seal-welded lids.

The seal welding of the canister lids and vent port covers utilizes multipass closure welds. These welds prevent the formation of possible leak paths by which the helium may escape. There is the possibility that a pinhole defect could be present in a given weld pass; however, by using multiple weld passes, the chance that there could be pinhole defects on successive weld layers that are all in alignment is considered to not be credible (Transnuclear, Incorporated, 2006). Thus, the canister is fabricated using multilayer, double-seal welds at each end, and multilayer circumferential and longitudinal welds of the canister shell are used to prevent leakage of the helium backfill.

### **1.2.2.2 Drying**

To begin the sealing process, the vacuum drying system is connected to the canister and a liquid pump is used to drain approximately 227 L [60 gal] of water from the canister, which lowers the water level about 102 mm [4 in] below the bottom of the shield plug. The automatic welding process begins on the top inner cover plate. During this process, the empty space is monitored for hydrogen buildup. If hydrogen is detected, the welding process is stopped. Transnuclear, Incorporated (2006) specifies the hydrogen concentration limit to be 2.4 percent. If this limit is exceeded, the cavity below the top inner plate is purged with helium at a pressure of 13,789–20,684 Pa [2–3 psig] until the hydrogen concentration falls below the 2.4 percent safety limit. Once this has been accomplished, the welding process of the inner seal weld is continued.

The canister is initially emptied of the bulk of the water by “blow down” using pressurized nitrogen, helium, or shop air fed through the vent tube, and the water is forced out the siphon tube. The canister then undergoes the standard procedure of a vacuum process to remove any residual liquid water and water vapor. Specifically, the pressure is reduced in steps of approximately 100, 50, 25, 15, 10, 5, and 3 torr [1.93, 0.97, 0.48, 0.29, 0.19, 0.10, and 0.06 psi]. After pumping down to each level, the pressure is monitored, and when the pressure stabilizes, the next vacuum process is restarted to complete the vacuum drying process. It may be

necessary to repeat some steps, depending on the rate and amount of the pressure increase during the hold period. Following the standard guidance, the vacuum drying is considered complete when the pressure stabilizes for a minimum of 30 minutes at 3 torr [0.06 psi] or less.

Once the canister pressure has stabilized, the canister is backfilled with helium and reevacuated a second time, which ensures that any remaining reactive gases are less than 0.25 percent by volume. After the second evacuation, the canister is backfilled with helium and slightly pressurized. Transnuclear specifies the canister pressurization to be approximately  $1.65 \times 10^5$  Pa [24 psia] but not to exceed  $2.34 \times 10^5$  Pa [34 psia]. At this point, a helium leak test of the inner canister lid seal weld is performed. If a helium leak is found, weld repairs are made, the canister is repressurized, and the helium leak test is repeated. Once no leaks are detected, the canister is depressurized.

At this point, the canister is reevacuated. As before, the vacuum levels are reduced in steps of approximately 10, 5, and 3 torr [0.19, 0.10, and 0.06 psi]. At each step, the vacuum process is stopped and the canister pressure is monitored until the pressure re-stabilizes. The second vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 torr [0.06 psi] or less. Once the leak test has been completed, helium feed lines are removed, the siphon and vent ports are seal welded closed, and a dye penetrant test is performed. The final step in the sealing process is to install the canister outer top cover plate using a second seal weld between the cover plate and the canister shell. This seal weld along with the inner seal weld provides the redundant sealing at the upper end of the dry shield canister. Note that the lower end of the canister also has redundant seal welds that are installed and tested during fabrication. The pressure boundary of the dry shielded canister is formed by the canister shell and the welded redundant cover plates. Because there are no penetrations in the pressure boundary, and thereby no leakage path, no pressure monitoring equipment is used. Once the dry shielded canister is welded and sealed, there is total confinement of the radioactive materials (Transnuclear, Incorporated, 2006).

Note that the vacuum drying system and the use of an automatic welding system are not considered important to safety. This is because their performance is not required to “provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public” (Transnuclear, Incorporated, 2006). Specifically, the failure of any of these systems may result in operational delay; however, this will not result in a hazard to the public or operating personnel, and therefore these components need not comply with the requirements of 10 CFR 72 (Transnuclear, Incorporated, 2006). Note, however, that the use of redundant welds, as provided by the inner and outer cover plates, is required as part of the 10 CFR 72 license (Transnuclear, Incorporated, 2006).

### **1.2.3 NAC-MPC<sup>7</sup>**

The NAC-MPC has two configurations. The Yankee-MPC designed to store up to 36 intact Yankee Class spent fuel assemblies, and the CY-MPC, designed to store up to 26 Connecticut Yankee fuel assemblies, reconfigured fuel assemblies, and damaged fuel in CY-MPC damaged fuel cans.

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<sup>7</sup>MPC—Multi-purpose canister.



### 1.2.3.1 Welding

The NAC-MPC fuel basket contains disks fabricated from aluminum alloy that are used for heat transfer from the spent fuel through the basket structure. At temperatures above 66 to 71 °C [150 to 160 °F], it is possible to produce a hydrogen concentration in the canister that approaches or exceeds the lower flammability limit for hydrogen of 4 percent (NAC International, 2004). Initially, there will be some clearance between the inner shield lid and the canister which could allow any reactive gases to escape. However, because the size of the gap cannot be consistently determined, the possibility of the collection of these gases under the shield lid must be considered. In addition, gas may collect under the lid because approximately 189 L [50 gal] of water is drained from the canister to allow welding operations to take place. Therefore, a monitoring system that can detect hydrogen at a concentration of 2.4 percent (NAC International, 2004) is required. This limit corresponds to 60 percent of the lower flammability limit of 4.0 percent (i.e.,  $0.6 \times 4.0 = 2.4$  percent). The hydrogen monitoring system is required to detect hydrogen prior to initiation of the welding process and also through the completion of the initial root pass of the weld. Any detection of hydrogen exceeding the concentration limit of 2.4 percent will require the welding operation to stop. As described in the final safety analysis report, if hydrogen is detected below the lid, a vacuum pump is connected to the vent port. The pump is then operated for a sufficient time period such that at least five times the air volume in the void space is removed by drawing ambient air through the gap between the shield lid and the canister shell (NAC International, 2004). This should ensure that any combustible gas concentrations are completely removed or sufficiently diluted. The hydrogen detector remains connected and is used to maintain a negative pressure in the canister. This will cause air to be drawn into the canister around the circumference of the weld and will prevent any additional hydrogen collection from occurring. Once the root pass weld is completed, the remaining weld passes will not be exposed to the inside atmosphere of the partially drained canister and the hydrogen monitoring process can be discontinued (NAC International, 2004). The root pass weld is to be examined using the dye penetrant weld method.

Once the inner shell lid is welded to the canister wall, NAC International specifies that the weld is to be examined using dye penetrant inspection of the final weld pass. The Yankee-MPC is drained of 189 L [50 gal] of water or the CY-MPC is drained of 246 L [65 gal] of water. The Yankee-MPC canister is then pressure tested (air over water) at  $1.03 \times 10^5$  Pa [15 psig], and the CY-MPC canister is tested at  $1.38 \times 10^5$  Pa [20 psig]. This pressure is held fixed for 10 minutes; any loss of pressure is unacceptable, and the leak must be located and repaired. After releasing the pressure, the weld is to be visually inspected for any indications of leaks or defects and a liquid penetrant examination of the final weld surface is to be performed.

### 1.2.3.2 Drying

At this point, the canister is drained of the bulk of the remaining water using a suction pump and, using nitrogen, the canister is pressurized to  $1.03 \times 10^5$  Pa [15 psig] and the remaining water is blown out of the drain port. The NAC final safety analysis report outlines a stepped vacuum drying process, similar to the previous canisters, for attaining the target 3 torr [0.06 psi] vacuum.

For the Yankee-MPC, holding the vacuum at  $\leq 3$  mm [3 torr] for a minimum of 30 minutes is used to verify that no water remains in the canister. Subsequently, two cycles of backfilling with helium at a pressure of  $1.0 \times 10^5$  Pa [1.0 atm] and re-evacuating to 3 mm [0.06 psi] of mercury are performed and the canister is helium leak tested using a mass spectrometer. If water

remains in the canister, the pressure will rise as the water vaporizes; however, the pressure should not continuously rise during the period of the test.

To perform the leaktight test, the leak test apparatus is used to create a volume above the shield lid that is evacuated. Using a mass spectrometer type of helium detector, this evacuated volume is tested for the presence of helium. A leaktight condition for the CY-MPC is achieved if a helium leak rate less than  $2 \times 10^{-7} \text{ cm}^3/\text{s}$  [ $1.22 \times 10^{-8} \text{ in}^3/\text{sec}$ ] (helium) is achieved using a leak test sensitivity of  $1 \times 10^{-7} \text{ cm}^3/\text{s}$  [ $6.1 \times 10^{-9} \text{ in}^3/\text{sec}$ ] (helium). The Yankee-MPC is considered leaktight if a helium leak rate less than  $8 \times 10^{-8} \text{ cm}^3/\text{s}$  [ $4.88 \times 10^{-9} \text{ in}^3/\text{sec}$ ] (helium) is achieved using a leak test sensitivity of  $4 \times 10^{-8} \text{ cm}^3/\text{s}$  [ $2.44 \times 10^{-9} \text{ in}^3/\text{sec}$ ] (helium).

Because the canister is vacuum dried and backfilled with helium after sealing, no significant moisture or other gases, such as air, remain in the canister. As a result, NAC International believes that there is no potential that radiolytic decomposition could cause an increase in canister internal pressure or result in a buildup of explosive gases in the canister (NAC International, 2004).

Once the inner canister lid has passed the leak test, the outer structural lid is welded to the canister shell. The first pass of the weld is inspected using a liquid penetrant test, and subsequent weld passes are also inspected using a progressive liquid penetrant or ultrasonic test examination. NAC International does discuss the possible repressurization of the canister. As per the operating procedure mentioned previously, the canister is vacuum dried and backfilled with helium at approximately one atmosphere prior to installing and welding the vent and siphon port covers. While in service, the internal pressure is expected to increase due to an increase in helium temperature and due to the assumed failure of fuel rod cladding. The cladding failure is estimated to be approximately 3 percent of the fuel rods, and these fuel rod failures would release 30 percent of the available fission gases in those rods. In the CY-MPC for breached fuel, the fuel rods are assumed to have lost both fission and rod backfill gas prior to loading in the canister and therefore do not contribute to canister pressure.

Additional discussion of maximum allowable vacuum drying times for the NAC-MPC can be found in Section 3.

## **1.2.4 Holtec HI-STAR 100 System**

### **1.2.4.1 Welding**

The welded confinement vessel of the HI-STAR 100 system is referred to as the MPC. The MPC provides radionuclide confinement under normal, off-normal, and accident conditions. The MPC itself consists of the canister shell, a bottom baseplate, and a redundant closure system at the top of the canister. All MPC pressure boundary welds are subjected to helium-leak-rate testing. Note that because the canister shell and the baseplate are treated as a single component of the containment boundary, the MPC shell and baseplate are helium-leakage tested during fabrication. Once the spent fuel has been loaded, the MPC lid, vent, and drain port cover plates and a closure ring are installed providing a redundant closure system. In the HI-STAR 100 system, the MPC lid is designed with a sufficient thickness to minimize radiation exposure to workers during MPC closure welding operations (Holtec International, Incorporated, 2007a).

Following the sequence of the closure process, the vent and drain port cover plates are welded to the MPC lid following completion of draining, drying, and the helium backfill. These cover plate welds are also helium-leak tested. The vent and drain ports are equipped with metal-to-metal seals to minimize leakage and to withstand the long-term effects of temperature and radiation (Holtec International, Incorporated, 2007a). However, because the vent and drain ports are covered by plates that are welded to the MPC lid, no credit is taken for the seals provided by the vent and drain ports (Holtec International, Incorporated, 2007a). No other confinement penetrations exist in the MPC. Thus, the MPC is a totally seal-welded pressure vessel, and because a welded redundant closure system is utilized, no direct monitoring of the closure is required. However, in the event of off-normal or accident conditions, there are requirements for verifying the confinement capabilities of the MPC. A helium gas sample is taken from the overpack (which envelopes the MPC in a helium environment and effectively forms the helium retention boundary). If the gas sample contains radioactive gas, the confinement boundary of the MPC is considered to have been breached. In this case, the overpack is backfilled with helium to the pressure specified for the MPC. The overpack serves as the confinement boundary.

With respect to the welding process during closure operations, Holtec International, Incorporated (2007a) discusses the possibility of hydrogen generation. According to Holtec International, Incorporated (2007a), there are numerous variables (e.g., aluminum particle size, pool temperature, pool chemistry) that dictate the extent of the hydrogen produced, making it very difficult to predict the amount of hydrogen that may be generated during MPC loading or unloading. Therefore, due to the variability in hydrogen generation, operating procedures require monitoring for combustible gases when a possible ignition source is present; that is, during welding or cutting operations (Holtec International, Incorporated, 2007a). The remedy for the presence of these gases is to either exhaust or purge the void space under the MPC lid during loading and unloading operations. Once the MPC is drained, dried, and backfilled with helium, the source of hydrogen gas is no longer present.

Holtec specifies that “appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC cutting operations” (Holtec International, Incorporated, 2007a). Specifically, the void space below the MPC lid is to be purged with an inert gas before and during MPC cutting operations to provide additional assurance that flammable gas concentration will not develop in this space.

The helium-leakage tests on the MPC closure welds are performed using a helium mass spectrometer leak detector with a specified minimum test sensitivity of  $2.5 \times 10^{-6} \text{ cm}^3/\text{s}$  [ $1.53 \times 10^{-7} \text{ in}^3/\text{sec}$ ] (helium) (Holtec International, Incorporated, 2007a), and the confinement boundary welds are required to have leakage rates not exceeding  $5 \times 10^{-6} \text{ cm}^3/\text{s}$  [ $3.1 \times 10^{-7} \text{ in}^3/\text{sec}$ ] (helium) (Holtec International, Incorporated, 2007a). In addition, the lid-to-shell weld is hydrostatically tested. The hydrostatic testing of the MPC confinement boundary is specified to be 125 percent of the design pressure [ $6.88 \times 10^5 \text{ Pa}$  [100 psig]] and to have a hold period of 10 minutes. Subsequent to the hold period, while maintaining the  $8.62 \times 10^5 \text{ Pa}$  [125 psig] pressure, the lid-to-shell weld is to be visually examined for any leakage and reexamined using the dye penetrant process. Evidence of any flaws such as cracking or deformation will be cause for rejection, or repair and retest (Holtec International, Incorporated, 2007a).

For the MPC lid-to-shell weld, both the root and final passes are examined by multilayer (where “layer” refers to each weld layer) liquid penetrant or volumetrically examined. If the volumetric-type examination method is selected, ultrasonic testing is the specific method to be

used (Holtec International, Incorporated, 2007a). In addition, if volumetric examination is used, then liquid penetrant examinations of only the root and final weld passes are performed. However, if volumetric examination is not used, then a multilayer liquid penetrant examination is performed on the lid-to-shell weld and, as specified by Holtec, the multilayer liquid penetrant examination must, at a minimum, include the root and final weld layer and one intermediate examination after every approximately 8.5 mm [3/8 in] of weld depth has been completed (Holtec International, Incorporated, 2007a). The 8.5-mm [3/8-in] weld depth corresponds to the maximum allowable flaw size in the weld (Holtec International, Incorporated, 2007a). As part of the evaluation of any weld flaws as identified by nondestructive examination, consideration of any active flaw mechanisms are also addressed. For example, if cyclic loading of the lid-to-shell weld is not significant, fatigue will not be an issue. In addition, the weld is protected from the external environment by the closure ring while the root of the weld is dry and inert (due to the helium atmosphere), making stress corrosion cracking a nonissue (Holtec International, Incorporated, 2007a).

Holtec states that a number of errors would have to happen for the MPC lid weld to fail:

- (i) improper weld by either a welding machine or a welder failing to use approved procedures;
- (ii) failure of a qualified inspector performing liquid penetrant inspections to detect a weld flaw;
- (iii) failure to detect a weld flaw from the volumetric inspections;
- (iv) failure to detect a leak in violation of the leakage rate during the hydrostatic test; and finally
- (v) failure of the leakage detection equipment itself (Holtec International, Incorporated, 2007a).

Even if the MPC lid seal weld fails, redundant sealing is provided by the closure ring. Therefore, consequences of a lid seal weld failure are that the closure ring will maintain the integrity of the confinement boundary. The complete confinement boundary is defined by the sealed, cylindrical MPC shell that is welded to a solid baseplate, the lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell (Holtec International, Incorporated, 2007a).

#### **1.2.4.2 Drying**

There are two possible approaches to dry the MPC internals: vacuum drying and forced helium dehydration drying. The vacuum drying process is specified for low-burnup fuels, while the forced helium dehydration process is specified for a canister that contains at least one high-burnup fuel assembly (Holtec International, Incorporated, 2007b). The forced helium dehydration process may also be used for moderate burnup fuels.

For the vacuum drying process, the MPC is evacuated to a pressure of less than or equal to 3 torr [0.06 psi] and held at this pressure for a minimum of 30 minutes. Holtec International, Incorporated (2007b) also specifies that the MPC is to be vacuum dried using a stepped process to prevent the formation of ice in the MPC and in the vacuum lines. However, specifics of the stepped process Holtec International, Incorporated (2007b) used were not given in the safety analysis report.

In the forced helium dehydration process, warm, dry helium is circulated through the MPC. The helium maintains the fuel in a cooled state while moisture is being removed from the MPC by the warm and dry helium. The forced helium dehydration process is a closed loop system. The moist helium leaves the MPC and is circulated through the demister, which removes the absorbed water by cooling the gas. The now-dry gas is circulated back through the MPC. By preheating the helium before it enters the MPC, the rate of moisture removal can be accelerated. The MPC is considered to be dry when the temperature of the gas exiting the

demoisturizer is  $\leq -6$  °C [21 °F] for  $\geq 30$  minutes (Holtec International, Incorporated, 2007a,b) such that the partial pressure of the water vapor in the MPC is less than 3 torr [0.06 psi].

As Holtec specified in their HI-STORM 100 system, for canisters containing at least a single high-burnup fuel assembly, the forced helium dehydration method for drying should be used to meet the normal peak cladding temperature limit and to satisfy the 65 °C [149 °F] variation temperature cycle criterion (Holtec International, Incorporated, 2007b). Holtec specifies that the forced helium dehydration method provides fuel cooling during the moisture removal process through forced convective heat transfer. The forced helium dehydration process results in a state of forced convection heat transfer in the canister as opposed to the natural mode of convection present in long-term storage. Therefore, it may be concluded that the peak fuel cladding temperature under storage conditions will be greater than that during the forced helium dehydration operation phase. In addition, heat transfer due to convection will maintain the fuel cladding temperature below the peak cladding temperature limit specified for normal conditions of storage. The cladding temperature should remain well below the high-burnup cladding temperature limit of 400 °C [752 °F] for all combinations of spent nuclear fuel type, burnup, decay heat, and cooling time (Holtec International, Incorporated, 2007b).

Following drying, the MPC shall be backfilled with 99.995 percent minimum purity helium. To ensure there is adequate heat transfer from the fuel to the fuel basket and surrounding structure of the MPC requires having the proper helium backfill pressure (Holtec, International Incorporated, 2007a). Providing a helium pressure {approximately  $1.5 \times 10^5$  to  $1.93 \times 10^5$  Pa [22 to 28 psig] (Holtec International, Incorporated, 2007a)} greater than atmospheric pressure at room temperature 21 °C [70 °F], prevents air in-leakage into the MPC because the cavity pressure rise due to helium gas heat up caused by fuel decay heat during storage. Recall that the helium leak rate is measured after each welding (sealing) step to determine whether the fuel is adequately confined.

### **1.3 Summary**

In this report, the safety analysis reports for four different spent fuel canister systems were reviewed. Items of interest were primarily limited to methods of sealing (i.e., welding of the canisters) and the procedures for drying the canisters. Drying and backfilling with helium limits the amount of oxidizing gases in the MPC below the target value of 0.25 percent by volume, which is necessary to prevent the oxidation of the spent fuel. In addition, the helium backfill provides for heat transfer during storage, provides an inert atmosphere to maintain the spent fuel integrity, and provides the means of leak-rate testing of the canister confinement boundary welds.

The spent fuel canister is a seal-welded pressure vessel with no penetrations and is designed to withstand the maximum internal pressure for all design basis conditions. In addition, because the canister is seal welded, no monitoring of the interior of the canister is required. Multipass welds prevent leakage issues associated with the helium fill gas used to inert the canister. The review found that the sealing methods for all canisters rely on redundant, multi-pass welds. By using multipass welds, the likelihood of a flaw made during a single weld pass, such as a pinhole, aligning itself with other possible weld defects in subsequent layers and completely penetrating the total thickness of the weld is considered not to be credible.

The drying procedures reviewed specified either the common vacuum drying method or the forced gas (helium) dehydration method. For the vacuum drying process, the vendors recommended using a stepped process to prevent the formation of ice in the canister and the

vacuum system lines. At each step, the length of the hold time at each pressure level depended on the rate and amount of the expected pressure increase due to the vaporization of water and other volatiles. The dryness criterion for the vacuum process is that the vacuum remains at 3 torr [0.06 psi] or less for a period of 30 minutes. This criterion appears to have become the industry standard for vacuum drying.

In this review, Holtec was the only vendor that briefly discussed the forced gas dehydration method and it was noted that this method must be used when high-burnup fuel (even if only a single fuel assembly) is to be stored. The gas (i.e., helium) aids in the heat transfer within the canister and helps to maintain fuel cladding temperature. For the forced gas dehydration process, when the temperature of the gas exiting the demister is  $\leq -6$  °C [21 °F] for  $\geq 30$  minutes, the interior of the canister is considered to be dry.

Table 1-1 summarizes the different drying processes and some of the criteria for determining whether the sealing/welds are considered to be leaktight. In some cases, details of the leaktight criteria or the helium pressures for the leaktight tests were not found in the safety analysis reports. Note that the Holtec canister also performs a hydrostatic pressure test on their canister in addition to the helium leaktight test.

| <b>Table 1-1. Summary of Drying and Sealing Criteria</b> |  |  |  |
|--|--|--|--|
|  | <b>Drying Process</b>  | <b>Sealing Pressure/Leaktight Test Criteria</b>  | <b>Limitations/Concerns Drying</b>   |
| NAC: Yankee-MPC  | Vacuum   | Leaktight (helium):<br>$4 \times 10^{-8}$ cm <sup>3</sup> /s [ $2.4 \times 10^{-9}$ in <sup>3</sup> /s]  | Possible formation of ice in canister  |
| NAC: CY-MPC  | Vacuum   | Leaktight (helium):<br>$2 \times 10^{-7}$ cm <sup>3</sup> /s [ $1.22 \times 10^{-8}$ in <sup>3</sup> /s]   |  |
| Transnuclear   | Vacuum   | $1.65 \times 10^5$ Pa [24 psia] but not to exceed $2.34 \times 10^5$ Pa [34 psia] (helium)<br>Leaktight criteria not specified   |  |
| FuelSolutions  | Vacuum   | 86,184.5 Pa [12.5 psig] canister pressure (helium)/<br>$8.52 \times 10^{-6}$ cm <sup>3</sup> [ $5.2 \times 10^{-7}$ in <sup>3</sup> /s]  |  |
| Holtec   | Vacuum (low burnup)<br>Forced helium dehydration (moderate to high-burnup) | 125 percent of the design pressure {869,475 Pa [100 psig]} (air over water hydrostatic test)/<br>Leaktight (helium): $5 \times 10^{-6}$ cm <sup>3</sup> /s [ $3.05 \times 10^{-7}$ in <sup>3</sup> /sec] | Forced helium dehydration depends on proper gas circulation throughout canister and basket (i.e., flow stagnation) |

## 2 HIGH-BURNUP FUEL ISSUES

### 2.1 Introduction

As given by Interim Staff Guidance (ISG)–11 (NRC, 2003a), a number of issues need to be considered with regard to high-burnup fuel (i.e., fuel with burnups exceeding 45 GWd/MTU). One issue that appears to be of significant concern is the deleterious effect of oxides or hydrides that can form on the cladding walls which causes thinning of the wall.

Three primary aspects affecting cladding conditions during vacuum drying are: (i) number of drying cycles, (ii) drying time per cycle, and (iii) the estimated spent fuel temperature during a cycle. In addition, it is also necessary to account for the time required to perform the initial sealing of the canister (i.e., welding of the lid shield plug). It is most likely that the cladding temperature cannot be measured during these operations, and therefore estimates must be made using analytical modeling.

The processes of drying and backfilling (with an inert gas) a canister may be classified as a “short-term” operation as opposed to storage conditions that are considered long-term. For such short-term operations, ISG–11 (NRC, 2003a) recommends that the “maximum calculated fuel cladding temperature” should not exceed a limit of 400 °C [752 °F]. In addition, ISG–11 (NRC, 2003a) specifies that thermal cycling (i.e., heatup/cooldown cycles) is permitted with no limit for temperature variations less than 65 °C [117 °F]. However, it should be limited to less than 10 cycles for cladding temperature variations greater than 65 °C [117 °F] for each cycle (NRC, 2003a).

### 2.2 Irradiation Hardening and Creep

As a result of irradiation hardening, cladding ductility is expected to decrease with increasing burnup leading to a decrease in creep. Therefore, it could be inferred that creep information available from dry storage studies for burnups lower than 45 GWd/MTU could possibly be applied to higher burnup cladding. However, higher stresses due to higher fission gas release, thinner cladding due to oxidation and hydride irradiation reduce the failure strain. Also, if the cladding experiences sufficiently high temperatures (quite possibly in the drying process), any irradiation hardening could be annealed out, which would lead to higher creep rates than expected (Jain, et al., 2004). Jain, et al. (2004) show that for a fuel rod with high burnup (67 GWd/MTU), a significant fraction of the irradiation hardening may undergo annealing at temperatures above 420 °C [788 °F] in a matter of hours to days. As has been discussed previously, with regard to creep, the following conclusions can be made: (i) deformation caused by creep will decrease the fuel rod pressure over time and (ii) a decrease in the cladding temperature will also lead to a decrease in rod pressure (and therefore the hoop stress), which would in turn decrease the corresponding creep rate. Exposure to high temperatures could potentially result in annealing of the fuel cladding material, thus reducing the cladding strength (i.e., yield strength and elastic modulus) to that of the unirradiated condition.

### 2.3 Hydride Formation

For high-burnup fuel, the susceptibility for hydride formation is of concern because of (i) higher hydrogen content, (ii) increased corrosion (wall thinning), and (iii) higher fission gas pressure, which in turn leads to a higher cladding stress.

Daum, et al. (2006) determined that typical high-burnup Zircaloy-4 fuel cladding contains up to 600–800 weight parts per million of hydrogen. At room temperature, the hydrogen primarily precipitates in the form of circumferentially oriented hydrides that localize in the form of a radially localized layer rim. During short-term drying operations in which temperatures may be equal to or greater than 400 °C [752 °F], it is likely that greater than 200 weight parts per million of hydrogen will be in solution and the hydrogen can precipitate as radial hydrides upon cooling (Daum, et al., 2006). However, to nucleate these hydrides along the radial direction requires a critical cladding circumferential stress, at elevated temperatures, that is near the threshold. If this stress is lower than this threshold value, only circumferential hydride precipitation occurs.

Drying operations could possibly make the (Zircaloy-4) fuel cladding more susceptible to failure during fuel handling and possible post-storage retrieval (Daum, et al., 2006). Specifically, hydride precipitates that are initially present in the circumferential direction may orient themselves to the radial direction of the cladding if the hoop stress exceeds a given stress level at or above a corresponding threshold temperature.

Pool transfer and subsequent drying operations involve thermomechanical processes that are believed to lead to the formation of radial hydrides. These radial hydrides may lead to reduced cladding ductility and impact resistance and affect the claddings capability to withstand subsequent handling operations, especially any hypothetical accident conditions.

Daum, et al. (2006) indicated that the threshold stress is approximately 75 to 80 MPa [ $1.09 \times 10^4$  to  $1.16 \times 10^4$  psi] for both unirradiated and high-burnup stress-relieved Zircaloy-4 fuel spent fuel cladding. When the cladding was cooled from 400 °C [752 °F] and subjected to ring compression tests at both room temperature and 150 °C [302 °F], it was determined that radial-hydride precipitation embrittles Zircaloy-4. Moreover, the plastic tensile hoop strain that is required to initiate unstable crack propagation along radial hydrides decreased from 8 percent to less than 1 percent as the fraction of radial hydrides increased.

Current licensing guidelines specify that the peak cladding temperature should not exceed 400 °C [752 °F] during normal short-term operations such as from-pool transfer, drying, and backfilling with an inert gas. As specified in ISG-11 (NRC, 2003a), for low burnup fuel ( $\leq 45$  GWd/MTU), a higher temperature limit of 570 °C [1,058 °F] is allowed during short-term operations “if it can be shown by calculation that the estimated cladding hoop stress is equal to or less than 90 MPa [13,053 psi] for the temperature limit proposed”. These temperature and stress limits are based upon the concern for radial hydride formation. According to Daum, et al. (2006), the effect of hydrides on the cladding could be exacerbated for high-burnup fuel because of the cladding’s higher hydrogen content. In addition, hydride formation could lead to possible degradation in mechanical performance and susceptibility to cladding failure (Daum, et al., 2006).

There are two situations in which hydride reorientation can occur in the cladding: during vacuum drying and during long-term storage which is characterized by a slowly decreasing temperature. As presented in the Electric Power Research Institute (2000) report, laboratory experiments were performed on test specimens subjected to repeat thermal cycling with a simultaneously applied high tensile stress. These experiments were conducted to study hydride reorientation from the circumferential to radial direction. It was determined that, in addition to tensile stresses, the reorientation was highly dependent on the number of thermal cycles. The Electric Power Research Institute (2000) reported that tens of thermal cycles were required to produce any significant hydride reorientation. However, it was concluded that the number of thermal cycles which cladding may be subjected to during typical vacuum drying process are



minimal in comparison with those required in the experimental test, and therefore the cumulative effect of thermal cycling can be disregarded (Electric Power Research Institute, 2000).

As discussed previously, because the cladding threshold stress limit for hydride reorientation may be exceeded, the presence of some radial hydrides following vacuum drying may occur (Electric Power Research Institute, 2000). One approach is to limit the drying temperature, which could reduce the number of circumferential hydrides that dissolve and then reprecipitate in the radial direction during cooling. There are two ways in which the reorientation of hydrides interacts with the stress in the cladding. First, the hydrides may interlink and form long radial hydrides (Electric Power Research Institute, 2000). The second interaction involves the way in which the reorientation of hydrides interacts with preexisting cladding cracks or notches in the cladding. The presence of a biaxial stress field at a crack tip may attract hydrides, causing them to concentrate in the zone also around the crack tip (Electric Power Research Institute, 2000).

## **2.4 Delayed Hydride Cracking**

The suitability for dry storage of spent fuel is controlled by the condition of the cladding. The potential loss of cladding integrity can be caused by the following damage mechanisms when considering the dry storage life: tertiary creep (a major damage mechanism) is characterized by an increasing creep rate eventually leading to rupture, delayed hydride cracking, and stress corrosion cracking. These failure mechanisms are controlled by the initial state of the cladding, which is strongly dependent on the thermomechanical and physical state of the cladding before being placed in dry storage (Electric Power Research Institute, 2000). Two conditions are necessary for delayed hydride cracking to occur: the availability of hydrogen in excess of the solid solubility (i.e., the degree to which one solid component can dissolve another to form a solid solution) and the cladding temperature history.

In zirconium alloys (i.e., zircaloy cladding), hydrogen embrittlement may occur for extended fuel burnup, which in turn may lead to significant reduction in fracture toughness. In addition to gross embrittlement, hydrides are responsible for a delayed failure mode that is characterized by a slow evolution of failure. Specifically, hydrogen diffuses in the material and forms hydrides when the terminal solid solubility is exceeded. The hydride is a brittle phase and may cause delayed hydride cracking—a subcritical crack growth mechanism in which the crack propagates slowly, eventually leading to failure. It is generally accepted that the process of delayed hydride cracking includes stress induced hydrogen diffusion, hydride precipitation, and hydride fracturing. One important characteristic of the temperature history that promotes delayed hydride cracking is the possibility of exceeding the upper temperature limit during vacuum drying followed by a continuously decreasing temperature. In addition, thermal cycling due to repetition of vacuum drying and backfill operations may be a contributing factor. However, initiation of delayed hydride cracking will not occur unless the appropriate stress condition is satisfied (i.e., if there is sufficient stress to activate the delayed hydride cracking mechanism), regardless of any other promoting mechanism (Electric Power Research Institute, 2000). The origin of these stresses will not be discussed in this in this report.

## **2.5 Oxide Layer Thickness**

As given in ISG-11 (NRC, 2003a), high-burnup fuel may have cladding walls that have become relatively thin from in-reactor formation of oxides or hydrides. When the structural integrity of the cladding is evaluated, “the applicant should specify the maximum cladding oxide thickness and the expected thickness of the hydride layer (or rim)” (NRC, 2003a). Specifically, cladding

stress calculations should use an effective cladding thickness that is reduced by the thickness of the oxide and hydride layers.

The value of cladding oxide thickness is required to be verified by the use of oxide thickness measurements, experimentally validated computer codes using measured oxide thickness data, or alternative methods that the NRC finds appropriate (NRC, 2003a). It is also noted that oxidation may not be of a uniform thickness along the axial length of the fuel rods (NRC, 2003a).

For high-burnup fuel (up to 60 GWd/MTU), the FuelSolutions W21 canister requires that the cladding oxide layer thickness for storing fuel assemblies is verified to not exceed 70  $\mu\text{m}$  [ $2.76 \times 10^{-3}$  in]. Although the formation of hydrides in the cladding increases with increasing oxide layer thickness, hydriding itself does not appear to significantly reduce the ductility of high-burnup cladding (EnergySolutions Spent Fuel Division, Incorporated, 2007). This oxide layer thickness is determined by a measurement of a statistical sample of the fuel assemblies, and for an oxide thickness greater than 70  $\mu\text{m}$  [ $2.76 \times 10^{-3}$  in], the fuel is considered damaged (EnergySolutions Spent Fuel Division, Incorporated, 2007). This is true for boiling water reactor fuels only.

## 2.6 Summary

Understanding various mechanisms that may affect cladding integrity is of concern at all burnup levels. These mechanisms arise from the thermomechanical processes to which cladding is subjected.

Creep has been considered as a self-limiting process. Specifically, as the cladding creeps the internal pressure (i.e., hoop stress) decreases, leading to a decrease in creep rate. The mechanism of irradiation hardening may lead to a decrease in cladding ductility where the amount of decrease is expected to be dependent on the level of burnup. This loss of ductility would lead to a decrease in the amount of creep. However, it is also possible that the cladding irradiation hardening could be annealed out. As discussed in this section, it has been determined that a significant amount of annealing could occur in a matter of hours to days; thus, it is quite possible that annealing could occur during the drying process of sufficiently high temperatures are used. The end result of annealing would be an increase in the creep rate.

For high-burnup fuel one issue that appears to be of significant concern is formation of radial hydrides in the cladding. The formation of radial hydrides may lead to degradation in mechanical properties and an increased susceptibility to cladding failure. However, orienting the initially circumferential hydrides to the radial direction requires a critical cladding circumferential stress near the threshold at elevated temperatures. It has been shown that the reorientation is highly dependent on the number of thermal cycles. However, as stated previously, tens of cycles are required for this reorientation to take place. For example, this combination of stress and temperature could occur during drying operations. In connection with the formation of hydrides is the effect of delayed hydride cracking. However, as noted previously for radial hydride formation, the appropriate stress state must also be present for delayed hydride cracking to take place. For high-burnup fuels, thinning of cladding walls will occur due to the formation of oxides. Thus, any cladding stress calculations should account for this thinning by using an effective thickness.

This discussion shows the importance of the physical state and thermomechanical history of the cladding, which can lead to a number of sometimes competing cladding damage mechanisms.

## **3 THERMAL OPERATION LIMITS FOR DRYING PROCESS**

### **3.1 Introduction**

In accordance with NUREG-1536 (NRC, 1997), water inside the MPC cavity during wet transfer operations is not permitted to boil. By preventing the water inside the canister from boiling, possible uncontrolled pressure on the cask and the dewatering and inerting systems is avoided and discharge of liquids that may be providing radiation shielding is prevented (NRC, 1997). Boiling is prevented by limiting the amount of time the spent fuel can be submerged in water after the canister is removed from the pool but before the beginning of the drying operation. Holtec specifies that the users of the HI-STAR system must have a procedure in place to monitor the elapsed time from the removal of the canister from the spent fuel pool until the beginning of the canister drying process to prevent water boiling in the canister.

### **3.2 Temperature Distribution Under Vacuum Conditions**

As noted previously, the canister is loaded in the spent fuel transfer pool and is subsequently drained, dried, and backfilled with helium. For the HI-STAR type of multipurpose canisters, the vacuum drying technique used involves evacuating the canister for a period of time to remove the remaining moisture that is present after drying. As in most spent fuel loading operations, the canister is contained within an overpack (e.g., a shielded transfer cask with the annular gap between the canister and the overpack filled with water). This water blanket helps maintain the fuel cladding temperatures below the design basis limits. During the draining operation, the heat-generating spent fuel is progressively uncovered and the spent fuel basket and the spent fuel itself will gradually heat up. (Initially cool conditions were present while the basket and fuel were submerged in the water contained within the canister.) Heat transfer from the spent fuel basket to the canister shell occurs by conduction and radiation. Therefore, the annular gap between the canister and the overpack must be kept filled with water during the draining and drying process.

#### **3.2.1 Effect of Fuel Cladding Crud Resistance**

As part of the thermal analysis of the canister and spent fuel, the amount of crud deposits needs to be accounted for. The composition of the crud is primarily iron oxides with small amounts of other materials; therefore, the heat conductivity of the crud can be approximated by the range of conductivities of typical metal alloys. This approximation is deficient because the crud has pores inside, unlike a metal alloy. In a representative thermal analysis performed by Holtec International, Incorporated (2007a), it was concluded that the formation of crud does not significantly change the spent nuclear fuel cladding temperature.

#### **3.2.2 Maximum Time Limit During Wet Transfer**

In Holtec International, Incorporated (2007a), the thermal inertia of the complete HI-STAR system is reported to limit the temperature rise. That is, the mass of the water, canister, fuel assembly, and the overpack will absorb the decay heat produced by the spent fuel. This should ensure a slow temperature increase starting from the initial temperature of the HI-STAR system (Holtec International, Incorporated, 2007a).

Table 3-1 shows the maximum allowable time to complete the wet transfer operations. Should the maximum allowable time to complete the wet transfer operations be exceeded, Holtec operating procedures specify that a forced water recirculation be employed. In this process, cool water is introduced into the canister through the drain port in the canister lid with the heated water being extracted through the vent port.

| <b>Table 3-1. Maximum Allowable Time Duration for Wet Transfer Operations*</b> |                   |
|--|-------------------|
| Initial Temperature °C [ °F]   | Time Duration (h) |
| 46 [115]   | 46.7              |
| 49 [120]   | 44.3              |
| 52 [125]   | 41.9              |
| 54 [130]   | 39.5              |
| 57 [135]   | 37.1              |
| 60 [140]   | 34.6              |
| 63 [145]   | 32.3              |
| 66 [150]   | 29.8              |

\*Holtec International, Incorporated. "Final Safety Analysis Report for the Holtec International Storage, Transport and Repository Cask System (HI-STAR 100 Cask System)." Rev. 3. Marlton, New Jersey: Holtec International, Incorporated. May 2007.

Holtec International, Incorporated (2007a) shows the minimum water flow rate required to maintain the canister water temperature below boiling with an adequate subcooling margin to be a function of the decay heat load, the water heat capacity, the maximum canister water mass temperature, and the temperature of the water supply.

The NAC-MPC Final Safety Analysis Report (NAC International, 2004) specifies time at temperature durations for their MPC system. For this case, the canister can be filled with water for a maximum of 22 to 33 hours (after removal from the spent fuel pool) before the start of the vacuum drying process with an initial water temperature of 38 °C [100 °F] (NAC International, 2004). NAC specifies that if the time limit is not met for the vacuum condition, the application of in-pool or forced air cooling is necessary so that the fuel cladding and basket component temperatures do not exceed their short-term allowable temperatures. Specifically, NAC technical specifications require that in-pool cooling of the transfer cask be initiated and maintained for a minimum of 24 hours with the canister backfilled with helium (NAC International, 2004). Alternatively, forced air cooling of the transfer cask may be utilized. In this case, air with a maximum temperature of 24 °C [75 °F] is supplied to the transfer cask annulus fill/drain lines at a rate of 10.6 m<sup>3</sup>/s [375 ft<sup>3</sup>/min] for a minimum of 24 hours with the canister filled with helium (NAC International, 2004).

NAC has placed the following limits for the vacuum drying times: (i) the time duration from completion of draining through completion of testing for vacuum dryness and the introduction of helium backfill shall not exceed 16 hours, and (ii) the time duration from the end of external forced air cooling or in-pool cooling until the completion of testing for vacuum dryness and also the introduction of the helium backfill shall not exceed 10 hours (NAC International, 2004). In the event that these time limits cannot be met, NAC International has specified the following procedure: (i) fill with helium (completion time, 2 hours); (ii) place the transfer cask containing the helium-filled loaded canister into the spent fuel pool (completion time, 2 hours); and (iii) maintain the transfer cask in the spent fuel pool for a minimum of 24 hours. Note that an alternative to (iii) is the use of the forced air cooling process.

### **3.3 Summary**

This section briefly discussed maintaining a sufficiently cool temperature to prevent boiling of the water inside the canister. Boiling may not only cause uncontrolled pressure on the cask but also the possible discharge of liquids that provide shielding.

Very limited information was found in the four safety analysis reports reviewed in Section 1. Thus, the previous discussion was limited to the Holtec HI-STAR and NAC-MPC systems. Holtec specifies a 30- to 47-hour time limit for wet transfer operations. If this time is exceeded, Holtec employs a forced water recirculation method in which cool water is introduced into the canister. NAC specifies a time duration of 22 to 33 hours before the start of the vacuum drying process, after which either forced air cooling or a return to the spent fuel pool is used to provide cooling of the canister. Either cooling method requires a minimum duration of 24 hours. NAC also provided information on vacuum drying time limits. A time limit of 16 hours is specified— from the completion of draining through the completion of vacuum drying—while a time limit of 10 hours is used when vacuum drying follows any supplemental cooling that was used to prevent boiling.

## 4 OVERALL SUMMARY

Regulations of 10 CFR Parts 71 (transportation) and 72 (storage) govern handling of spent nuclear fuel. One important requirement of 10 CFR Part 72 regulations is the retrievability of the spent fuel. Because of this, preservation of the spent fuel cladding in particular is a significant concern [10 CFR 72.112(h)(1)].

Cladding during reactor operations is subjected to radiation-induced hardening, hydrogen pickup, and oxidation, which increases with the level of burnup and may affect the structural integrity of the fuel rods. Appropriate vacuum drying methods and leak testing procedures are described in NUREG-1536 (NRC, 1997). Drying is necessary to reduce or eliminate the amount of water in the canister, and leak testing ensures that the dry inert atmosphere is maintained. The proper closure weld design, in which ISG-15 (NRC, 2001) gives guidance, is also important in maintaining an inert atmosphere such that cladding oxidation is minimized. Therefore, the requirement of preserving cladding integrity, as well as confinement and containment considerations, is what drives the selection of particular methods of drying and sealing.

The following sections are structured in the form of a checklist of topics pertaining to drying and sealing that may be considered when reviewing a safety analysis report.

### 4.1 Drying

- Drying method
  - Vacuum drying: evaluate how the applicant determines the possible formation of ice in the canister will be addressed (i.e. a stepped vacuum process). Check target dryness criteria {e.g., 3 torr [0.06 psi] for 30 minutes}.
  - Forced helium dehydration: evaluate how the applicant will ensure the proper circulation of gas throughout the canister internal structure. Check target dryness criteria.
- Drying cycle considerations
  - Expected total drying cycle time.
  - Peak temperatures and their effect on spent fuel cladding integrity (e.g., possible hydride formation), irradiation hardening, hydrogen embrittlement, and creep.
  - Evaluate how the applicant will determine contingencies for supplemental cooling if specified time limit is exceeded.

### 4.2 Sealing/Welding

- Leak testing
  - Evaluate how the applicant will ensure that the process conforms to ANSI N14.5 and the allowable leakage rate.

- Evaluate how the applicant will determine whether redundant sealing systems will solely rely on welds or work in combination with other means such as metal seals.
- Evaluate how the applicant will establish precautions for possible hydrogen build-up during root pass welding of inner top lid to canister shell that requires a monitoring system and purge of void space.
- Evaluate how the applicant will determine whether the canister will be subjected to pressure testing (e.g., hydrostatic air-over-water) select the target test pressure to be used and the form of inspection method to verify weld integrity.
- Welding issues
  - Check that the applicant specifies the type of welding to be used (i.e., gas tungsten arc welding, gas metal arc welding, or other method).
  - Check that the applicant specifies the form of nondestructive examination to be used: liquid dye penetrant, frequency of penetrant testing (i.e., multilayer), or volumetric testing (specifically, an ultrasonic form of test).
- Evaluate how the applicant will determine the maximum allowable flaw size and its basis (i.e., ASME code based, fracture/stress analysis).

### **4.3 Other Issues**

- Time-to-boil analysis should consider the time from in-pool removal to completion of first seal weld (inner lid to canister body) and remaining draining of canister water.
  - Evaluate whether the applicant is aware of possible over-pressurization of canister.
  - Ascertain cladding temperature limits.
- Determine that the applicant will purge the canister with 99.995 percent purity helium.
  - Check that the applicant specifies the helium backfill density.
  - Check that the applicant specifies the method for limiting residual gas concentration to 0.25 vol% to reduce amount of oxidants.
- Check quality assurance program, which is important for proper canister welding and nondestructive evaluation.

### **4.4 Current Transportation, Aging, and Disposal Container Requirements**

Based upon the discussion that has been presented in the previous sections, a quick look at the current design requirements for the proposed transportation, aging, and disposal canister follow:

- Transportation, aging, and disposal canister design is to satisfy either of these criteria:

- Closure welds shall meet ISG–18 (NRC, 2003b) for assuring no credible leakage.
- A leak-tight condition must be demonstrated, which is based upon ANSI N14.5.
- Helium shall be used for final backfill.
- Limit maximum oxidizing gas concentration to 0.20 percent of the free volume of the transportation, aging and disposal canister.
  - Sample fill gas to verify purity.

For draining, drying, and backfill operations, cladding temperature shall not exceed 570 °C [1,058 °F]. Note that the last item specifies that for short-term conditions, the allowable temperature is given as 570 °C [1,058 °F]; this limit is applicable to low to moderate burnup spent fuel. For high-burnup fuel, the limit is 400 °C [752 °F].

ISG–15 (NRC, 2001) provides reasonable assurance no credible leakage can occur. Also, if ISG–15 (NRC, 2001) guidelines are followed, it can be concluded that no undetected flaws of significant size will exist.



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